UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

September 15, 2000

EA-99-220

Charles M. Dugger, Vice President
Operations - Waterford 3
Entergy Operations, Inc.
17265 River Road
Killona, Louisiana 70066-0751

SUBJECT: NRC INSPECTION REPORT NO. 50-382/00-08 AND EXERCISE OF ENFORCEMENT DISCRETION

Dear Mr. Dugger:

This refers to the inspection conducted on July 2 through August 19, 2000, at the Waterford Steam Electric Station, Unit 3, facility. The enclosed report presents the results of this inspection. The results of the implementation of the permanent plant modification program were discussed on July 21, 2000, with you and other members of your staff. A supplemental exit meeting was conducted with Mr. E. Perkins and other members of licensee management by telephone on August 1, 2000, to discuss the closure of Unresolved Item 50-382/9915-01. The remainder of the results of this inspection were discussed on August 24, 2000, with you and other members of your staff.

This inspection was an examination of activities conducted under your licenses as they relate to safety, compliance with the Commission's rules and regulations, and with the conditions of your licenses. Within these areas, the inspection consisted of a selective examination of procedures and representative records, observations of activities, and interviews with personnel.
The NRC is exercising enforcement discretion in accordance with Section VII.B.6 of the NRC's Enforcement Policy and refraining from issuing a violation for a Severity Level IV violation of 10 CFR 50.59 (EA 99-220). The issue involved the automatic resequencing of nonsafety loads to the Class 1E bus following a diesel generator start, and would be an unreviewed safety question under the current rule. Discretion was warranted because the same issue would not be a violation under the revised 10 CFR 50.59 rule (64 FR 53582).

Based on the results of this inspection, three issues of very low safety significance (green) were identified. These issues were determined to involve violations of NRC requirements. However, the violations were not cited because of their very low safety significance and because they have been entered into your corrective action program. If you contest these noncited violations, 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.

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/

Linda Joy Smith, Chief
Project Branch E
Division of Reactor Projects

Docket No.: 50-382
License No.: NPF-38

Enclosures:
NRC Inspection Report No.
50-382/00-08

cc w/enclosures:
Executive 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
One American Place, Suite 1630
Baton Rouge, Louisiana 70825-1697

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

Ronald Wascom, Administrator
and State Liaison Officer
Louisiana Department of Environmental Quality
P.O. Box 82215
Baton Rouge, Louisiana 70884-2215

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

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

Electronic distribution from ADAMS by RIV:
Regional Administrator **(EWM)**
DRP Director **(KEB)**
DRS Director **(ATH)**
Senior Resident Inspector **(TRF)**
Branch Chief, DRP/E **(LJS)**
Senior Project Engineer, DRP/E **(GAP)**
Branch Chief, DRP/TSS **(LAY)**
RITS Coordinator **(NBH)**

Only inspection reports to the following:
David Diec **(DTD)**
NRR Event Tracking System **(IPAS)**
WAT Site Secretary **(AHY)**
Dale Thatcher **(DFT)**
G. F. Sanborn, D:ACES **(GFS)**
K. D. Smith, RC **(KDS1)**
R. W. Borchardt, OE **(RWB1)**
OE:EA File **(RidsOeMailCenter)**
G. M. Vasquez (GMV)

DOCUMENT NAME: R:_WAT\2000\WT2000-08RP-TRF.wpd

RIV:RI:DRP/E	SRI:DRP/E	RI:DRS/EMB	RI:DRS/EMB	RI:DRS/EMB
JMKeeton	TRFarnholtz	RPMullikin	CEJohnson	MLRunyan
T-TRFarnholtz for	***T-LJSmith***	***/RA/***	***/RA/***	***/RA/***
09/14/00	09/14/00	09/14/00	09/14/00	09/14/00
C:EMB	ACES	AC:DRP		
JLShackelford	GFSanborn	LJSmith		
/RA/	***/RA/***	***/RA/***		
09/14/00	09/15/00	09/15/00		

OFFICIAL RECORD COPY T=Telephone E=E-mail F=Fax

ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.: 50-382

License No.: NPF-38

Report No.: 50-382/00-08

Licensee: Entergy Operations, Inc.

Facility: Waterford Steam Electric Station, Unit 3

Location: 17265 River Road
Killona, Louisiana

Dates: July 2 through August 19, 2000

Inspectors: T. R. Farnholtz, Senior Resident Inspector
J. M. Keeton, Resident Inspector
C. E. Johnson, Senior Reactor Inspector
R. P. Mullikin, Senior Reactor Inspector
M. F. Runyan, Senior Reactor Inspector

Approved By: L. J. Smith, Chief, Project Branch E

ATTACHMENTS:

Attachment 1: Supplemental Information

Attachment 2: NRC's Revised Reactor Oversight Process

SUMMARY OF FINDINGS

IR 05000382-00-08; on 07/02-08/19/00; Entergy Operations Inc.; Waterford 3; Integrated Resident & Regional Report; Permanent Plant Modification, Postmaintenance Testing, Other (Problem Identification and Resolution).

The inspection was conducted by resident inspectors and regional reactor inspectors. This inspection identified three green findings, all of which were noncited violations, and one finding of no color. The significance of issues is indicated by their color (green, white, yellow, or red) and was determined by the significance determination process.

Cornerstone: Mitigating Systems

- Green. The inspectors identified during a review of Permanent Plant Modification ER-W3-99-0857-00-00 and previous test records that Shutdown Cooling Header Thermal Relief Valve S-404A failed its bench test and exceeded its design set point by greater than 22 percent on October 6, 1995. The licensee reset Valve SI-404A to within design limits; however, the licensee failed to initiate a condition report for this condition adverse to quality to identify the root cause and apparent condition that may have existed on other relief valves. The failure to initiate a condition report upon discovery of this condition adverse to quality was a violation of 10 CFR Part 50, Appendix B, Criterion XVI and Site Procedure W2.501, "Corrective Action." This violation is being treated as a noncited violation in accordance with Section VI.A of the NRC Enforcement Policy and is in the licensee's corrective action program as Condition Report CR-WF3-2000-0822.

  This issue was characterized as a "green" finding using the significance determination process. It was determined to have a very low risk significance because even though the valve exceeded its design set point, sufficient margin existed to maintain the integrity of the piping protected by the valve. The licensee reset the valve at the time of discovery to its design set point, and the licensee has since tested the valve and found the as-found set point satisfactory. (Section 1R17).

- Green. On three occasions personnel failed to enter the appropriate Technical Specification limiting condition for operation when equipment was unable to perform its intended safety function. The plant stack wide range gas monitor and containment isolation Valve CS-129A were rendered inoperable to perform maintenance and the fuel handling building crane failed a surveillance test. In each case, the components should have been declared inoperable and the provisions of the applicable Technical Specification should have been entered. These errors were three examples of a violation of Technical Specification 6.8.1.a. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. These errors were placed in the corrective action program as Condition Reports 2000-0765, -0777, and -0785.

  The inspectors assessed the risk significance of these errors using the reactor safety significance determination process. The inspectors found that the issue had very low risk significance because the provisions of the applicable Technical Specification actions were met by default in each case (Section 1R15).

- Green. Three examples of inadequate corrective action were identified related to main control board switch knob replacement. The switches were associated with a containment isolation valve, a boric acid makeup pump recirculation valve, and a boric acid makeup pump. This event is a repeat of two similar events identified in 1999 where similar knobs were replaced without assuring that the control circuit design was not altered. Corrective actions taken following the 1999 events failed to prevent recurrence. The failure to establish effective corrective actions to prevent recurrence of improperly installed control switch knobs, a significant condition adverse to quality, was a violation of 10 CFR Part 50, Appendix B, Criterion XVI. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy in the corrective action program as Condition Report 2000-0770.

  The inspectors assessed this issue using the reactor safety significance determination process. The inspectors found that the issue had very low risk significance because the valves downstream of the containment isolation valve were closed and the boric acid system components would have gone to their safe condition if a safety injection actuation signal is generated (Sections 1R19 and 4OA5.2).

- No Color. During a previous inspection, the NRC inspectors identified an unresolved item involving a potential violation of 10 CFR 50.59 concerning the automatic resequencing of nonsafety loads to the Class 1E bus following a diesel generator start. The Updated Final Safety Analysis Report indicated that nonsafety loads were only reintroduced manually under administrative controls. This issue was determined to be a violation of 10 CFR 50.59 and constituted an unreviewed safety question. However, it was determined that this issue would not be a violation under the revised 10 CFR 50.59 rule, currently scheduled to be effective January 2001. This judgement is based on the conclusion that the change did not represent more than a minimal increase in the probability of a malfunction of equipment important to safety. Therefore, in accordance with Section 8.1.3 of the NRC Enforcement Manual (NUGEG/BR-0195, Revision 3), enforcement discretion was exercised after consultation with the Office of Enforcement pursuant to Section VII.B.6 of the NRC Enforcement Policy and a violation was not issued (EA-99-220).

  The inspectors found that the issue had very little safety significance because the nonsafety loads had at least single breaker protection and were not ordinarily vulnerable to faulted conditions (Section 4OA5.1).

Report Details

Summary of Plant Status: The plant was operating at approximately 100 percent power at the beginning of this inspection period and remained at that level for the entire inspection period.

1 REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity (R)

1R04 Equipment Alignments (71111.04)

1. Containment Spray Pump B Alignment

a. Inspection Scope

The inspectors reviewed the mechanical and electrical alignment of Containment Spray Pump B, which was lined up in standby while Containment Spray Pump A was taken out of service for maintenance. The review was conducted using Operating Procedure OP-009-001, "Containment Spray," Revision 10.

b. Issues and Findings

There were no findings identified during this inspection.

.2 Alignment of High-Pressure Safety Injection Train B

a. Inspection Scope

The inspectors reviewed the mechanical and electrical alignment of High-Pressure Safety Injection Train B, which was in standby alignment while High-Pressure Safety Injection Train A was removed from service for scheduled maintenance. The inspectors also verified the availability of High-Pressure Injection Train AB, which was available but required manual alignment to place in service. The review was conducted using Operating Procedures OP-009-008, "Safety Injection System," Revision 15, and OP-903-030, "Safety Injection Pump Operability Verification," Revision 13.

b. Issues and Findings

There were no findings identified during this inspection.

.3 Complete Walkdown of Emergency Diesel Generator A Standby Alignment

a. Inspection Scope

The inspectors performed a complete walkdown of the mechanical and electrical alignment of Emergency Diesel Generator A. The inspectors verified the correct systems alignment in accordance with operating procedures, including abnormal and emergency, the Updated Final Safety Analysis Report, and drawings. The systems walkdown included valve positions, electrical alignment, component labeling, material condition of components and systems, and essential support systems. The walkdown was performed using Operating Procedure OP-009-002, "Emergency Diesel Generator," Attachments 11.1 and 11.3, Revision 17, and Plant Drawings LOU-1564-G-164, “Miscellaneous Reactor Auxiliary Systems,” Sheets 1 and 2.

b. Issues and Findings

There were no findings identified during this inspection.

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors conducted tours and assessed the material condition of the active and manual fire suppression systems in the following areas:

- Safeguards Pump Room A
- Emergency Diesel Generator Room B
- Electrical Switchgear Rooms A, B and AB on the +21-foot elevation of the reactor auxiliary building

b. Issues and Findings

There were no findings identified during this inspection.

1R11 Licensed Operator Requalification (71111.11)

a. Inspection Scope

The inspectors observed the conduct of a simulator scenario for a staff crew and witnessed the scenario critique. The crew was evaluated in the simulated control room using a scenario that had not been seen by the staff operators. The inspectors interviewed several operators with respect to use of the plant monitoring computer (nonsafety) indication during performance of emergency operating procedures. The inspectors discussed the reliance on plant computer indications with the Operations Branch to determine current NRC position on using nonsafety indications during emergency operating procedure implementation.

b. Observations and Findings

There were no findings identified during this inspection.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

The inspectors reviewed the maintenance rule data for the following to determine if the maintenance rule scope for these systems had been appropriate:

- Plant Monitoring Computer - The review considered the unrestricted use of the plant monitoring computer during routine plant operation and mitigation of accidents. This review had been prompted by the poor performance of the plant monitoring computer. The inspectors reviewed the maintenance history, interviewed reactor operators, and assessed the maintenance rule functions for this system.

- Core Protection Calculators - The review focused on several failures of core protection calculator channels and treatment of the unavailability time associated with the failures. The inspectors interviewed reactor operators, system engineers, and the maintenance rule coordinator, and reviewed the maintenance history and assessed the maintenance rule functions for this system.

- Essential Chill Water System B - The inspectors reviewed the maintenance history, interviewed the system engineer, and assessed the maintenance rule functions for this system.

b. Issues and Findings

There were no findings identified during this inspection.

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

.1 Maintenance on Containment Spray Pump A

a. Inspection Scope

The inspectors reviewed Work Control Packages MAI 411995, 408037, and 418422 to determine the extent of work planned and that appropriate risk assessments had been considered. The inspectors also reviewed the Technical Specifications to verify that the licensee was in compliance with these requirements throughout the duration of the maintenance activity.

b. Issues and Findings

There were no findings identified during this inspection.

.2 Failure of Reactor Coolant System Hot Leg Temperature Detector

a. Inspection Scope

The inspectors reviewed Work Control Package MAI 418110 and verified that the appropriate risk assessments had been performed during replacement of the temperature detector with an installed spare temperature detector. The inspectors also reviewed the Technical Specifications related to the affected core protection calculator to verify that the licensee was in compliance with these requirements throughout the duration of the maintenance activity.

b. Issues and Findings

There were no findings identified during this inspection.

.3 Failure of Essential Chiller A Control Module

a. Inspection Scope

The inspectors reviewed Work Control Packages MAI 406745 and MAI 419381 to verify that the appropriate risk assessments had been conducted prior to troubleshooting and replacement of the control module in the chiller control circuit. The inspectors also reviewed the Technical Specifications related to the affected essential chiller to verify that the licensee had been in compliance with these requirements throughout the duration of the maintenance activity.

b. Issues and Findings

There were no findings identified during this inspection.

1R15 Operability Evaluations (71111.15)

.1 Failure to Enter Appropriate Technical Specification Requirements

a. Inspection Scope

The inspectors reviewed the operability evaluations associated with three condition reports written during this inspection period to document failures to enter applicable Technical Specification limiting condition for operation.

b. Issues and Findings

On July 6, 2000, maintenance technicians deenergized the plant stack wide range gas monitor to replace a velocity flow probe. The monitor remained deenergized just over 8 hours to perform the maintenance. During this time, the provisions of Technical Specification 3.3.3.1 (radiation monitoring instrumentation) applied. However, the licensee failed to enter this Technical Specification action as required. The required actions were (1) restore the monitor to operable status within 72 hours or initiate a preplanned method for monitoring the plant stack and (2) restore the monitor to operable status within 7 days or prepare and submit a special report to the NRC. These actions were not required in this case since the monitor was inoperable for less than 72 hours. This event was documented in Condition Report 2000-0765. The operability evaluation associated with this condition report was adequate.

On July 12, the electrical breaker for Containment Isolation Valve CS-129A was opened during the performance of planned maintenance on the containment spray system. This had the effect of rendering this valve inoperable and the provisions of Technical Specification 3.6.3 (containment isolation valves) applied. However, the licensee failed to enter this Technical Specification action as required. The specified action was to isolate the subject penetration within 4 hours. The valve remained deenergized and closed during this maintenance activity effectively meeting this requirement. This event was documented in Condition Report 2000-0777. The operability evaluation associated with this condition report was adequate.

On July 12, during the performance of a surveillance test on the fuel handling building crane, it was identified that a limit switch was faulty. Upon discovery, the crane interlock system should have been declared inoperable and Technical Specification 3.9.7 (crane travel - fuel handling building) entered. However, the licensee failed to take these actions. The Technical Specification action required that the crane be placed in a safe position (not over irradiated fuel assemblies). The crane did remain in a safe position during this event. This event was documented in Condition Report 2000-0785. The operability evaluation associated with this condition report was adequate.

Technical Specification 6.8.1.a requires, in part, that licensees implement procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. The regulatory guide recommends administrative procedures for authorities and responsibilities for safe operation and shutdown. Section 5.1.6 of Operations Procedure OP-100-014, "Technical Specification and Technical Requirements Compliance," Revision 10, states "If any system, subsystem, or component becomes unable to perform its intended safety function due to surveillance, calibration, or maintenance, then declare that equipment inoperable and enter the appropriate Technical Specification/Technical Requirements Manual action." Contrary to this requirement, the licensee failed to enter the appropriate Technical Specification action as required on the three occasions described above. Failure to enter the appropriate Technical Specification actions could potentially result in the failure to take the appropriate compensatory actions required by Technical Specifications. These failures to enter the appropriate Technical Specification actions are identified as three examples of a violation of Technical Specification 6.8.1.a. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as Condition Reports 2000-0765, -0777, and -0785 (NCV 50-382/00008-01).

.2 Nonconservative Setpoint in the Shield Building Ventilation System

a. Inspection Scope

The inspectors reviewed the operability evaluation associated with Condition Report 2000-0809. This condition report was generated to document a condition in which the shield building ventilation system setpoint was determined to be nonconservative. The operability evaluation included a detailed discussion of the setpoint and the calculated conditions in the annulus following a loss-of-coolant accident.

b. Issues and Findings

There were no findings identified during this inspection.

1R16 Operator Work-Arounds (71111.16)

a. Inspection Scope

The inspectors reviewed selected operator workarounds and evaluated the effects on the operator's abilities to implement the required actions during routine and accident conditions. The inspectors also reviewed the cumulative effects of outstanding operator workarounds potential for causing system misoperation, degrading event mitigation capabilities, and timeliness impact on response to plant transients and accidents. The inspectors verified that the operators had been identifying workarounds in accordance with Operating Instruction OI-002-000, "Annunciator, Control Room Instrumentation and Workarounds Status Control," Revision 18.

b. Issues and Findings

There were no findings identified during this inspection.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

The inspectors reviewed procedures governing plant modifications to evaluate the effectiveness of the licensee's programs for implementing modifications to risk significant systems, structures, and components, such that these changes did not adversely affect the design and licensing basis of the facility. The inspectors also reviewed 20 permanent plant modification packages (7 design change documents, 11 engineering requests, and 2 plant change documents) to verify that they were performed in accordance with plant procedures. Procedures and permanent plant modifications reviewed are listed in the attachment.

The inspectors conducted field walkdowns of 10 permanent plant modifications, identified in the attachment. The cognizant design and/or system engineers for the identified modifications were interviewed as to their understanding of the modification packages.

The inspectors evaluated the effectiveness of the licensee's corrective action process to identify and correct problems concerning the performance of permanent plant modifications. In this effort, the inspectors reviewed condition reports and the subsequent corrective actions pertaining to licensee identified problems and errors in the performance of permanent plant modifications. Condition reports reviewed are listed in the attachment.

b. Issues and Findings

The inspectors determined that permanent plant modifications appeared to be implemented in accordance with licensee site procedures. The inspectors reviewed Engineering Request ER-W3-99-0857-00-00, "SI 404A and B Relief Valves Back Pressure Consideration," and requested current and previous test records for Shutdown Cooling Header Thermal Relief Valves SI-404A and 404B. Review of these test records indicated that Relief Valve SI-404A exceeded its design set point (2485 psig) by 22 percent on October 6, 1995. The licensee reset Valve SI-404A to within design limits; however, the licensee failed to initiate a condition report for this condition adverse to quality to identify the root cause and apparent condition that may have existed on other relief valves. Criterion XVI of 10 CFR Part 50, Appendix B, states, in part, that measures shall be established to assure that conditions adverse to quality, i.e., as failures, are promptly identified and corrected. Site Procedure W2.501, "Corrective Action," Section 4.9.3, required a condition report to be initiated when a condition adverse to quality, such as a failure of a component to meet a surveillance or postmodification test acceptance criteria is identified. Failure to initiate a condition report upon discovery of a condition adverse to quality is a violation of 10 CFR Part 50, Appendix B, Criterion XVI and Site Procedure W2.501, "Corrective Action." However, this condition is considered a noncited violation (NCV 50-382/00008-02), consistent with Section VI.A of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as Condition Report CR-WF3-2000-0822.

The inspectors evaluated this condition using the significance determination process and found it to be of very low risk significance, in part, because sufficient margin existed to maintain the integrity of the piping up to the as-found set point overpressure protection provided by the valve. The licensee reset the valve at the time of discovery to its design set point, and the licensee has since tested the valve and found the as-found set point satisfactory. The inspectors concluded the safety significance of this issue was very low (green).

1R19 Post Maintenance Testing (71111.19)

.1 Containment Spray Pump A

a. Inspection Scope

The inspectors reviewed the postmaintenance testing conducted on Containment Spray Pump A and associated components. Extensive planned maintenance had been performed on this equipment. The postmaintenance testing performed included a walkdown and inspection of the pump and associated piping and a VT-2 inspection of specified components.

b. Issues and Findings

There were no findings identified during this inspection.

.2 Core Protection Calculator C

a. Inspection Scope

The inspectors reviewed the postmaintenance testing conducted on Core Protection Calculator C following installation of the spare hot leg temperature detector. The temperature detector was replaced with the installed spare temperature detector in the reactor coolant system loop. The postmaintenance testing was conducted in accordance with Operating Procedure OP-903-001, “Technical Specification Surveillance Logs,” Revision 22.

b. Issues and Findings

There were no findings identified during this inspection.

.3 Control Room Normal Ventilation Train A

a. Inspection Scope

The inspectors reviewed the postmaintenance testing conducted on Control Room Normal Ventilation Train A and associated components following completion of MAI 414221. Maintenance performance problems had caused the planned maintenance outage to be extended. The postmaintenance testing was performed in accordance with Technical Procedure PE-004-026, “HVC-101 and HVC-102 Leak Test,” Revision 5, and Operating Procedure OP-903-119, “Secondary Auxiliaries Quarterly IST Valve Tests,” Revision 5.

b. Issues and Findings

There were no findings identified during this inspection.

.4 Essential Chiller A

a. Inspection Scope

The inspectors reviewed the postmaintenance testing conducted on Essential Chiller A and associated components following completion of MAI 419381. The postmaintenance testing was performed in accordance with the work instructions.

b. Issues and Findings

There were no findings identified during this inspection.

.5 Control Board Switches

a. Inspection Scope

The inspectors reviewed a situation concerning inadequate maintenance performed on three main control board switches. The maintenance was performed to replace the plastic knobs, which were identified as being susceptible to failure.

b. Issues and Findings

On July 10, 2000, the licensee replaced three plastic control switch knobs on the main control board switches associated with Valve BD-102A (steam generator blowdown inside containment isolation valve), Valve BAM-126B (Boric Acid Makeup Pump B recirculation valve), and Boric Acid Makeup Pump B. After completion of the maintenance, it was discovered that the switch functions had been altered such that all three switches had a push-to-trip or a push-to-actuate feature. None of these switches were originally designed to function in this way.

An increase in the number of failed control switch knobs had been noted over the last several years. The knobs failed by cracking or crumbling under normal use. The inspectors were concerned that a knob could fail in a way that prevents the associated switch from being operated, particularly during an accident scenario. The licensee determined that the cause of the failures was the use of a chemical cleaning solution used to clean the main control board. The use of this chemical caused the plastic to become brittle and subject to failure. To correct this condition, the licensee planned to replace the knobs for those switches used in the emergency operating procedures and the off-normal procedures. The three knobs replaced during this inspection period were done for this reason.

The licensee developed special tools and techniques to replace these knobs with the switches energized. The knob assembly consists of a plastic knob pressed onto a metal shaft with a spacer placed under the knob to prevent the knob from being pressed thereby breaking the contacts in the switch and tripping or actuating the associated component.

The postmaintenance testing specified to be performed following the knob replacement on all three of these switches was to ensure that the knob was properly seated. In addition, the work package for Valve BD-102A specified that operations personnel were to perform Operations Procedure OP-903-119, “Secondary Auxiliaries Quarterly In-Service Test (IST) Valve Tests,” Revision 5. These tests were performed and considered satisfactory and the switches returned to service.

During operations shift turnover later that day, it was discovered that the switch for Valve BD-102A did not function as expected. If the knob was pressed, the valve fully opened. Upon this discovery, the licensee checked the other two switches that had knobs replaced earlier in the day and determined that they also did not function as expected. Specifically, Valve BAM-126B closed if the knob was pressed and Boric Acid Makeup Pump B stopped if the knob was pressed.

Further investigation revealed that the switch contacts for Valve BD-102A had been damaged during the knob replacement activity resulting in this component not functioning in accordance with the wiring diagram. The inspectors were concerned with this condition because this valve was danger tagged to the closed position at the time of discovery to allow work to be performed on the steam generator blowdown system. Because of the damaged switch and the improperly installed knob, this valve unexpectedly went to the open position. No personnel were injured and no equipment damage occurred.

The inspectors considered the postmaintenance testing performed on all three of these switches to have been inadequate. The specified actions could not be taken effectively since the switch remained energized. Because of this, the electrical maintenance technicians could not manipulate the switch in any way to ensure that the knob was properly seated without affecting the associated component. The specified postmaintenance testing failed to identify the improperly installed knobs and the damaged switch for Valve BD-102A. The licensee generated Condition Report 2000-0770 to place this issue in the corrective action program.

In addition, the inspectors had concerns with corrective actions taken to correct similar conditions identified in 1999. This aspect of this event is detailed in Section 4OA5 (Other) of this report. The inspectors considered the inadequate postmaintenance testing to be part of inadequate corrective actions taken following these events.

1R22 Surveillance Testing (71111.22)

.1 Low-Pressure Safety Injection Pump B In Service Test Surveillance

a. Inspection Scope

The inspectors witnessed the performance of the low pressure safety injection pump testing and taking of vibration data. The test was performed in accordance with Surveillance Procedure OP-903-030, "Safety Injection Pump Operability Verification," Revision 13.

b. Observations and Findings

There were no findings identified during this inspection.

.2 Emergency Diesel Generator B Test

a. Inspection Scope

The inspectors observed portions of the Emergency Diesel Generator B surveillance test on July 24, 2000. The test was conducted in accordance with Operating Procedures OP-903-068, "Emergency Diesel Generator and Subgroup Relay Operability Verification," Revision 12, and OP-009-002, "Emergency Diesel Generator," Revision 17.

b. Issues and Findings

There were no findings identified during this inspection.

.3 High-Pressure Safety Injection Pump B In-Service Test

a. Inspection Scope

The inspectors observed a scheduled surveillance test of High Pressure Safety Injection Pump B. The test was conducted in accordance with Operating Procedure OP-903-030, "Safety Injection Pump Operability Verification," Revision 13. The inspectors also reviewed Attachment 10.4, "HPSI Pump B IST Data."

b. Issues and Findings

There were no findings identified during this inspection.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed Engineering Request ER-W3-00-0487-00-02, “Temporary Alteration to Re-wire RC ITE0122 HD to RC ITE0121 X.” The purpose of this temporary modification was to replace a failed hot leg temperature sensor with a useable hot leg temperature sensor. The temporary alteration was performed in accordance with Procedure UNT-005-004, “Temporary Alteration Control,” Revision 14, and the appropriate work instructions.

b. Issues and Findings

There were no findings identified during this inspection.

4 OTHER ACTIVITIES (OA)

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

The inspectors reviewed the performance indicator data for “Safety System Unavailability (SSU) - Emergency AC Power System.” This performance indicator is included in the mitigating systems cornerstone.

b. Issues and Findings

There were no findings identified during this inspection.

4OA5 Other

.1 (Closed) Unresolved Item 50-382/99015-01: Failure to identify an unreviewed safety question involving resequencing of nonsafety loads to 1E bus.

The inspectors determined that the licensee's failure to obtain Commission approval for an Updated Final Safety Analysis Report correction regarding the resequencing of nonsafety loads to a Class 1E bus following a diesel generator start was a violation of 10 CFR 50.59 and constituted an unreviewed safety question. However, it was determined that this issue would not be a violation under the revised 10 CFR 50.59 rule, currently scheduled to be effective January 2001. This judgement is based on the conclusion that the change did not represent more than a minimal increase in the probability of a malfunction of equipment important to safety. Therefore, in accordance with Section 8.1.3 of the NRC Enforcement Manual (NUGEG/BR-0195, Revision 3), enforcement discretion is being exercised after consultation with the Office of Enforcement pursuant to Section VII.B.6 of the NRC Enforcement Policy and a violation is not being issued (EA-99-220). The licensee placed this issue in their corrective action program as Condition Report 98-0763**.**

.2 Control Board Switches

a. Inspection Scope

The inspectors reviewed the events surrounding the failure to properly install three knobs on main control board switches with regard to problem identification and resolution (PI&R). The improper installation resulted in changing the function of the switches.

b. Issues and Findings

As described in Section 1R19.5 (postmaintenance testing) of this report, the licensee replaced the plastic knobs on three control switches because they were subject to failure. The knobs were not installed properly, which resulted in the introduction of a push-to-trip or a push-to-actuate feature that was not in the original design. The inspectors concluded that the specified postmaintenance testing was not adequate because it failed to identify the improperly installed knobs and failed to identify a switch damaged during the knob replacement procedure.

Section M2.1 of NRC inspection Report 50-382/99-20 describes a previous event concerning two other control board switch knobs installed incorrectly. These two switches were associated with the Essential Chiller B compressor and the Emergency Diesel Generator A output breaker. These two switches also had a push-to-trip or a push-to-actuate feature resulting from the improper installation of the knobs. The inspectors identified that the licensee failed to provide documented instructions for the replacement of the control board knobs that were adequate to prevent unintended control circuit modifications.

The licensee had generated Condition Reports 1999-0147 (Essential Chiller B compressor) and -0920 (Emergency Diesel Generator A output breaker) to place these concerns in their corrective action program. However, the corrective actions taken for these two conditions did not prevent unintended control circuit modifications when similar knobs were installed on July 10, 2000.

Appendix B, Criterion XVI of 10 CFR Part 50, "Corrective Action," states, in part, "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition." Contrary to this requirement, the corrective actions taken following the two 1999 events, which were considered significant conditions adverse to quality, were inadequate to preclude repetition. The failure to establish effective corrective actions to prevent recurrence of unintended control circuit modifications is identified as a violation. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement

Policy. This violation is in the corrective action program as Condition Report 2000-0770 (NCV 50-382/00008-03).

The inspectors determined that this issue did not lead to any measurable change in plant risk; therefore, the issue had very low safety significance. Two valves downstream of Containment Isolation Valve BD-102A were closed at the time of this event. Valve BAM-126B closes on a safety injection actuation signal and Boric Acid Makeup Pump B receives a start signal if a safety injection actuation signal is generated. These two actions are independent of the main control board switches. The issue did not meet the initial significance determination process screening and is green.

4OA6 Meetings

Exit Meeting Summaries

.1 The regional based reactor inspectors presented the permanent plant modification program inspection results to Mr. C. M. Dugger, Vice President, Operations, and other members of the licensee staff at the conclusion of the inspection on July 21, 2000, and a re-exit by telephone to inform the licensee that the unresolved item was now a noncited violation on July 27, 2000. 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.

.2 A supplemental exit meeting was conducted with Mr. E. Perkins and other members of licensee management by telephone on August 1, 2000, to discuss the closure of Unresolved Item 50-382/9915-01. The details of this closure are presented in Section 4OA5 of this report.

No proprietary information was identified. The licensee acknowledged the findings presented during the meeting.

.3 The resident inspectors presented the inspection results to Mr. C. M. Dugger, Vice President, Operations, and other members of licensee management at the conclusion of the inspection on August 24, 2000. 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.

ATTACHMENT 1

PARTIAL LIST OF PERSONS CONTACTED

Licensee

M. Brandon, Manager, Licensing
T. Brennan, Supervisor, Engineering
J. Burke, Engineer, Design Engineering
C. DeDeaux, Supervisor, Licensing
J. Douet, Manager, Operations
C. M. Dugger, Vice President, Operations
E. Ewing, General Manager, Plant Operations
R. Fili, Manager, Quality Assurance
C. Fugate, Manager, Technical Support
D. Gallodoro, Supervisor, Configuration Management
P. Gropp, Acting Director, Engineering
J. Holman, Manager, Events Assessments
J. Hunsaker, Manager, Site Support
T. Lett, Superintendent, Radiation Protection
D. Madere, Engineer, Licensing
G. Matharu, Supervisor, Engineering
B. Matthew, Manager, Engineering Support
J. O'Hern, Manager, Training and Emergency Planning
D. Ortego, Assistant Manager, Operations
E. Perkins, Jr., Director, Nuclear Safety Assurance
R. Peters, Manager, Corrective Action and Assessment
R. Putnam, Supervisor, Engineering
J. Reese, Supervisor, Engineering
J. Ridgel, Manager, Plant Maintenance
L. Rushing, Manager, System Engineering
B. Thigpen, Director, Planning and Scheduling
D. Viener, Engineer, Design Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-382/00008-01	NCV	Failure to enter appropriate Technical Specification requirements (Section 1R15).
50-382/00008-02	NCV	Failure to initiate a condition report upon discovery of a condition adverse to quality (Section 1R17).
50-382/00008-03	NCV	Failure to establish effective corrective actions to prevent reoccurrence of improperly installed control switch knobs (Section 4OA5).

Closed

50-382/00008-01	NCV	Failure to enter appropriate Technical Specification requirements (Section 1R15).
50-382/00008-02	NCV	Failure to initiate a condition report upon discovery of a condition adverse to quality (Section 1R17).
50-382/00008-03	NCV	Failure to establish effective corrective actions to prevent reoccurrence of improperly installed control switch knobs (Section 4OA5).
50-382/99015-01	URI	Failure to identify and unreviewed safety question involving resequencing of nonsafety loads to 1E bus (Section 4OA5).

LIST OF ACRONYMS USED

CFR	Code of Federal Regulations
DC	design change
ER	engineering request
NCV	noncited violation
NRC	Nuclear Regulatory Commission
PC	plant change
URI	unresolved item

DOCUMENTS REVIEWED

Procedures

OP-100-001, Rev. 17	Duties and Responsibilities of Operators on Duty
OP-100-007, Rev. 13	Shift Turnovers
OP-100-010, Rev. 14	Equipment Out of Service
OP-100-016, Rev. 2	EOP Change, Revision, Verification, and Validation Process
OP-100-017, Rev. 0	Administrative Procedure Emergency Operating Procedure Implementation Guide
OI-038-000, Rev. 0	Emergency Operating Procedure Operations Expectations
PV-OP-902, Rev. 7	Parameters Values Document
OP-902-000, Rev. 8	Standard Post Trip Actions
OP-902-001, Rev. 8	Reactor Trip Recovery

OP-902-002, Rev. 8 Loss of Coolant Accident Recovery

Design Changes, Engineering Requests, Plant Changes

DC-3493, “Stroke Time Speed Reduction of CCW Isolation Valves,” Revision 2

DC-3504, “Containment Sump Level Transmitter Replacement,” Revision 0

DC-3506, “Auxiliary Steam Test Connection for EFW Pump A/B,” Revision 3

DC-3542, “ CCW Temporary Chiller Water Piping Train Separation Boundary,” Revision 0

DC-3546, “Replacement of SUPS 3B-S,” Revision 1

DC-3555, “Station and Instrument Air Unloader Valve Unreliability,” Revision 0

DC-3529, “Remote Manual Operating Capability for CVC-209,” Revision 1

ER-W3-98-0237, “Replacement of Target Rock Pressure Regulating Valves in the Essential Air System,” Revision 2

ER-W3-98-0590, “ Potential to Void in Auxiliary Component Cooling Water System to Essential Chillers,” Revision 2

ER-W3-98-0714, “DC 3560 Upgrade of RCS Temperature Transmitters and RTD Elements,” Revision 2

ER-W3-98-0758, “Provide Details for Changing Slope of Three Inch Oil “Equalizing” Line, Utilized on the Terry Turbine Skid, so that the High Point is Located at the Governor (North) End and also Allow Continuous Venting and Defoaming of the Oil Drain/Equalizing Lines,” Revision 1

ER-W3-98-0869, “Upgrade HBC Unit to Meet EPRI PPPM for Valves SI-602A(B),” Revision 1

ER-W3-98-0899, “Design Change to Install RAS Bypass for Valves SI-602A(B),” Revision 2

ER-W3-98-1085, “Add Desicant Filler/Breathers to MFIV Actuator Hydraulic Reservoirs,” Revision 0

ER-W3-98-1086, “Replacement of the Atmospheric Dump Valves Plug Assemblies and Seat Rings,” Revision 0

ER-W3-98-1232, “ Solenoid Valve Replacement for CVC-401, RC-606, and IA-909,” Revision 1

ER-W3-99-0198, ”Modify Insulation on RC Hot Leg to Allow for Installation of MNSA Clamps and Future Inspection for Leakage,” Revision 1

ER-W3-99–0857, SI 404A & B Relief Valves Backpressure Consideration,” Revision 0

PC-8010, “Atmospheric Dump Valve Testing and Maintenance Improvements,” Revision 0

PC-8020, “Broad Range Gas Monitoring System Replacement,” Revision 0

Condition Reports

CR-WF3-1994-1026	CR-WF3-1996-0429	CR-WF3-1996-1088
CR-WF3-1996-1726	CR-WF3-1996-1807	CR-WF3-1997-1759
CR-WF3-1997-1795	CR-WF3-1998-0435	CR-WF3-1998-0516
CR-WF3-1998-0988	CR-WF3-1998-1246	CR-WF3-1999-0204

Engineering Procedures

W4.104, “ Engineering Request Process,” Revision 3

W4.105, “Engineering Request Response Implementation and Closeout,” Revision 2

DEPT-I-004, “Engineering Request Process Guide,” Revision 4

Work Authorizations

WA 01176446, “Perform Acceptance Test for design Change ER-W3-98-0899, “RAS-Bypass for Valves SI-602A(B)”

WA 01176914, “Need a Retest WA for EOS 99-0328 (SI-602B)”

WA 99003557, “DC-3557 Acceptance Test Train B”

ATTACHMENT 2

NRC'S REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) 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 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	**Radiation Safety**	**Safeguards**
•Initiating Events	•Occupational	•Physical Protection
•Mitigating Systems	•Public	
•Barrier Integrity		
•Emergency Preparedness		

To monitor these seven cornerstones of safety, the NRC used 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. 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 plan, as described in the Action Matrix.

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