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

February 7, 2008

William R. Brian, Vice President of Operations
Grand Gulf Nuclear Station
Entergy Operations, Inc.
P.O. Box 756
Port Gibson, MS 39150

SUBJECT: GRAND GULF NUCLEAR STATION - NRC INTEGRATED INSPECTION REPORT 05000416/2007005

Dear Mr. Brian:

On December 31, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Grand Gulf Nuclear Station facility. The enclosed integrated report documents the inspection findings, which were discussed on January 7, 2007, with you and other members of your staff.

The 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. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

This report documents five NRC identified and self-revealing findings of very low safety significance (Green). Two of these findings were determined to involve violations of NRC requirements; however, because of the very low safety significance and because they were entered into your corrective action program, the NRC is treating these findings as noncited violations (NCVs) consistent with Section VI.A of the NRC Enforcement Policy. If you contest 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 76011-4005; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington DC 20555-0001; and the NRC Resident Inspector at the Grand Gulf Nuclear Station facility.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) 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).

Sincerely,

/RA/

Michael C. Hay, Chief
Project Branch C
Division of Reactor Projects

Docket: 50-416
License: NPF-29

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Inspection Report 05000416/2007005
w/Attachment: Supplemental Information

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SUNSI Review Completed: _WCW_____ ADAMS: √ Yes No Initials: ___WCW
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RIV:SRI:DRP/C	SRI:DRP/C	C:SPE:DRP/C	C:DRS/EB1	C:DRS/PSB
AJBarrett	RLSmith	WCWalker	RLBywater	MPShannon
/RA/Electronic	/RA/Electronic	/RA/	/RA/	/RA/
2/7/08	2/7/08	1/30/08	2/1/08	2/5/08

C:DRS/OB	C:DRS/EB2	C:DRP/C		
RELantz	LJSmith	MCHay		
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2/5/08	2/5/08	2/7/08		

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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-416
Licenses:	NPF-29
Report No.:	05000416/2007005
Licensee:	Entergy Operations, Inc.
Facility:	Grand Gulf Nuclear Station
Location:	Waterloo Road Port Gibson, Mississippi 39150
Dates:	October 1 through December 31, 2007
Inspectors:	Andy Barrett, Senior Resident Inspector Douglas Bollock, Project Engineer Paul J. Elkmann, Emergency Preparedness Inspector Phillip Qualls, Reactor Inspector Donald L. Stearns, Health Physicist Eddie Uribe, NSPDP
Approved By:	Michael C. Hay, Chief Project Branch C Division of Reactor Projects

ENCLOSURE

CONTENTS

SUMMARY OF FINDINGS . . . . . -3-

1R04 Equipment Alignment . . . . . -6-
1R05 Fire Protection . . . . . -7-
1R06 Flood Protection Measures . . . . . -8-
1R07 Heat Sink Performance (71111.07A) . . . . . -9-
1R11 Licensed Operator Requalification (71111.11) . . . . . -9-
1R12 Maintenance Effectiveness . . . . . -10-
1R13 Maintenance Risk Assessments and Emergent Work Evaluation . . . . . -11-
1R15 Operability Evaluations . . . . . -12-
1R19 Post-maintenance Testing . . . . . -13-
1R22 Surveillance Testing . . . . . -13-
1EP1 Exercise Evaluation . . . . . -15-
1EP6 Drill Evaluation . . . . . -16-

RADIATION SAFETY . . . . . -17-

2OS1 Access Control to Radiologically Significant Areas . . . . . -17-
2OS2 ALARA Planning and Controls . . . . . -18-

OTHER ACTIVITIES . . . . . -21-

4OA1 Performance Indicator Verification . . . . . -21-
4OA2 Identification and Resolution of Problems . . . . . -23-
4OA3 Event Follow-up . . . . . -26-
4OA6 Meetings, Including Exit . . . . . -28-
4OA7 Licensee-Identified Violations . . . . . -29-

ATTACHMENT: SUPPLEMENTAL INFORMATION . . . . . -29-
KEY POINTS OF CONTACT A-1
LIST OF DOCUMENTS REVIEWED A-2

SUMMARY OF FINDINGS

IR05000416/2007005; 10/1/07 - 12/31/07; Grand Gulf Nuclear Station -- Integrated Resident and Regional Report; Surveillance Testing, ALARA Planning and Controls, Identification and Resolution of Problems, Event Follow-up.

This report covered a 3-month period of inspection by resident inspectors and Regional office inspectors. These inspection activities identified five findings, two of which were noncited violations. The significance of most findings is indicated by their color (Green, White, Yellow, or 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's 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 Criterion V, "Instructions, Procedures, and Drawings," of 10 CFR Part 50, Appendix B for the failure to demonstrate compliance with Technical Specification Surveillance Requirement 3.5.3.1 due to an inadequate surveillance procedure. The reactor core isolation cooling system is vented at the injection valve through a hard-piped drain with no visual means of detecting air in the system. The inspectors determined that the procedure failed to contain adequate acceptance criteria to qualitatively or quantitatively assess abnormal amounts of air in the reactor core isolation cooling system. The licensee entered this issue into their corrective action program as Condition Report CR-GGN-2007-03818.

  The finding was greater than minor because it affects the procedure quality attribute of the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using the Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 worksheet, the finding was determined to have a very low safety significance in that it did not result in the actual loss of the reactor core isolation cooling system, and was not potentially risk-significant due to external initiating events (Section 1R22).

  Cornerstone: Initiating Events

- Green. The inspectors identified a finding involving a loss of condenser vacuum caused by improper troubleshooting of the seal steam pressure controller. Specifically, the licensee failed to provide adequate work instructions and procedural limitations during troubleshooting of the seal steam pressure controller. As a result, the plant experienced a loss of condenser vacuum and a plant transient. The licensee entered this issue into their corrective action program as Condition Report CR-GGN-2007-04626.

The finding was more than minor because it was associated with the initiating events cornerstone attribute of equipment performance and affected the associated cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during power operations. Using the MC 0609, "Significance Determination Process," Phase 1 Worksheet, the inspectors concluded that a Phase 2 evaluation was required because the finding impacted both the initiating event and mitigating systems cornerstone. The inspectors performed a Phase 2 analysis using Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," of Manual Chapter 0609, "Significance Determination Process," and the Phase 2 Worksheets for Grand Gulf Nuclear Station. The inspectors assumed that only the power conversion system was affected and all other mitigating systems were available. Based on the results of the Phase 2 analysis, the finding was determined to have very low safety significance.

The cause of the finding was related to the human performance crosscutting component of decision making, in that the licensee failed to use conservative assumptions during troubleshooting activities and performed these activities without determining the validity of the troubleshooting instructions and identifying possible unintended consequences [H.1(b)] (Section 4OA3).

- Green. A self-revealing finding was identified involving the failure of a plant service water piping flange due to an improper flow control valve design modification. Specifically, the licensee failed to perform an adequate review of an engineering modification and the maintenance work orders did not have detailed installation instructions. As a result, the plant experienced a plant transient. The licensee entered this issue into their corrective action program as Condition Report CR-GGN-2007-05040.

  The finding was more than minor because it was associated with the initiating events cornerstone attribute of equipment performance and affected the associated cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during power operations. Using the Manual Chapter 0609, “Significance Determination Process,” Phase 1 worksheet, the finding was determined to have very low safety significance because the finding did not contribute to the likelihood that mitigating equipment or functions would not be available following a reactor trip. The cause of the finding was related to the human performance crosscutting component of work practices in that the responsible engineers failed to perform adequate self and peer checking during the development and review of the design modification to the plant service water flow control check valves [H.4(a)] (Section 4OA3).

- Green. A self-revealing finding was identified involving a loss of condenser vacuum caused by plant operators improperly removing a steam jet air ejector from service. Specifically, the licensee failed to isolate the steam jet air ejector from service as delineated in the system operating instructions. As a result, the plant experienced a loss of condenser vacuum and a plant transient. The licensee entered this issue into their corrective action program as Condition Report CR-GGN-2007-05676.

The finding was more than minor because it was associated with the initiating events cornerstone attribute of equipment performance and affected the associated cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during power operations. Using the MC 0609, "Significance Determination Process," Phase 1 Worksheet, the inspectors concluded that a Phase 2 evaluation was required because the finding impacted both the initiating event and mitigating systems cornerstone. The inspectors performed a Phase 2 analysis using Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," of Manual Chapter 0609, "Significance Determination Process," and the Phase 2 Worksheets for Grand Gulf Nuclear Station. The inspectors assumed that only the power conversion system was affected and all other mitigating systems were available. Based on the results of the Phase 2 analysis, the finding was determined to have very low safety significance.

The cause of the finding was related to the human performance crosscutting component of work practices in that the control room supervisor failed to ensure supervisory and management oversight of work activities such that nuclear safety is supported [H.4(c)] (Section 4OA2).

Cornerstone: Occupational Radiation Safety

- Green. The inspectors reviewed a self-revealing noncited violation of Technical Specification 5.4.1.a for the failure to provide a detailed work order package to perform vent and fill operations on a pressure transmitter. Specifically, the licensee did not provide appropriate instructions in a work order package to properly isolate pressure Transmitter 1N64N006B prior to opening the drain valve. Consequently, this resulted in the release of radioactive gas from the system and an unplanned and unintended exposure for two individuals involved in the work activity.

  The finding is more than minor because it is associated with the occupational radiation safety attribute of program and process and affected the cornerstone objective because it involved unplanned and unintended dose to two workers. Using the Occupational Radiation Safety Significance Determination Process, the inspectors determined that the finding was of very low safety significance because: it did not involve: (1) as low as reasonably achievable planning and controls, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. In addition, this finding has a cross-cutting aspect in the area of work control associated with work planning because the licensee failed to properly plan work activities by incorporating specific plant system details into the work order to allow the instrumentation and control technicians to properly drain a pressure transmitter [H.3(a)] (Section 2OS2).

B. Licensee-Identified Violations

A violation of very low safety significance which was identified by the licensee was reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. This violation and corrective actions are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Grand Gulf Nuclear Station (GGNS) began the inspection period at full rated thermal power. On October 19, 2007, reactor power was reduced to approximately 45 percent due to a loss of plant service water. The plant returned to full rated thermal power on October 20, 2007. On November 14, 2007, reactor power was reduced to approximately 90 percent for three days following reactor pressure and power oscillations caused by the electro-hydraulic control (EHC) system and subsequent EHC testing. After the testing was completed, power was reduced to 70 percent in order to set a new control rod pattern. The plant returned to full rated thermal power on November 18, 2007. On November 29, 2007, the plant reduced power to 95 percent for approximately six hours to cycle the EHC system turbine control valve pressure converters following an increase in the frequency of reactor pressure oscillations. On December 6, 2007, reactor power was reduced to 74 percent following a loss of condenser vacuum. The plant reached full power on December 7, 2007, and remained at or near full rated thermal power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment (71111.04)

.1 Partial System Walkdowns

a. Inspection Scope

The inspectors: (1) walked down portions of the four listed risk important systems and reviewed plant procedures and documents to verify that critical portions of the selected systems were correctly aligned; and (2) compared deficiencies identified during the walkdown to the licensee's Updated Final Safety Analysis Report (UFSAR) and corrective action program (CAP) to ensure problems were being identified and corrected.

- Reactor heat removal system Train A on October 15, 2007, during a planned reactor heat removal system Train B system outage.
- Standby liquid control system Train B on October 25, 2007, following system maintenance.
- Division III standby diesel generator on November 1, 2007, during a planned Division II standby diesel generator system outage.
- Fuel pool cooling and cleanup system Train A on November 15, 2007, following system maintenance.

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed four samples.

b. Findings

No findings of significance were identified.

.2 Complete System Walkdown

a. Inspection Scope

The inspectors: (1) reviewed plant procedures, drawings, the UFSAR, Technical Specifications (TS), and vendor manuals to determine the correct alignment of the standby service water system; (2) reviewed outstanding design issues, operator workarounds, and UFSAR documents to determine if open issues affected the functionality of the standby service water system; and (3) verified that the licensee was identifying and resolving equipment alignment problems.

Documents reviewed by the inspectors included:

- Procedure 04-1-01-P41-1, "System Operating Instruction - Standby Service Water System Revision 125
- System Piping Diagram M-1061, "Standby Service Water System"
- Engineering Request ER-2005-0303

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

Quarterly Inspection

The inspectors walked down the seven listed plant areas to assess the material condition of active and passive fire protection features and their operational lineup and readiness. The inspectors: (1) verified that transient combustibles and hot work activities were controlled in accordance with plant procedures; (2) observed the condition of fire detection devices to verify they remained functional; (3) observed fire suppression systems to verify they remained functional and that access to manual actuators was unobstructed; (4) verified that fire extinguishers and hose stations were provided at their designated locations and that they were in a satisfactory condition; (5) verified that passive fire protection features (electrical raceway barriers, fire doors,

fire dampers, steel fire proofing, penetration seals, and oil collection systems) were in a satisfactory material condition; (6) verified that adequate compensatory measures were established for degraded or inoperable fire protection features and that the compensatory measures were commensurate with the significance of the deficiency; and (7) reviewed the UFSAR to determine if the licensee identified and corrected fire protection problems.

- Auxiliary Building Stairwells
- Division I switchgear room (Room 1A208)
- Security Central Alarm Station
- Standby service water Train A pump house (Room 1M110)
- Standby service water Train B pump house (Room 2M110)
- Standby service water Train A valve room (Room 1M112)
- Standby service water Train B valve room (Room 2M112)

Documents reviewed by the inspectors included:

- Procedure EN-DC-161, “Control of Combustibles,” Revision 1
- Grand Gulf Nuclear Station Fire Pre-Plans, Revision 15
- Procedure 01-S-10-1, “Fire Protection Plan,” Revision 103
- Procedure 10-S-03-9, “Control of Fire Pre-Plans,” Revision 2

The inspectors completed seven samples.

1R06 Flood Protection Measures (71111.06)

.1 Semi-annual Internal Flooding

a. Inspection Scope

The inspectors: (1) reviewed the UFSAR, the flooding analysis, and plant procedures to assess seasonal susceptibilities involving internal flooding; (2) reviewed the UFSAR and CAP to determine if the licensee identified and corrected flooding problems; (3) inspected underground bunkers/manholes to verify the adequacy of (a) sump pumps, (b) level alarm circuits, © cable splices subject to submergence, and (d) drainage for bunkers/manholes; (4) verified that operator actions for coping with flooding can reasonably achieve the desired outcomes; and (5) walked down the four areas listed below to verify the adequacy of: (a) equipment seals located below the floodline, (b) floor and wall penetration seals, © watertight door seals, (d) common drain lines and sumps, (e) sump pumps, level alarms, and control circuits, and (f) temporary or removable flood barriers.

- November 3, 2007, Standby service water Train A pump house (Room 1M110)
- November 4, 2007, Standby service water Train B pump house (Room 2M110)
- November 5, 2007, Standby service water Train A valve house (Room 1M112)
- November 5, 2007, Standby service water Train B valve house (Room 2M112)

Documents reviewed by the inspectors are included in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07A)

a. Inspection Scope

The inspectors reviewed licensee programs, verified performance against industry standards, and reviewed critical operating parameters and maintenance records for the Division II standby diesel generator jacket water heat exchanger. The inspectors verified that: (1) performance tests were satisfactorily conducted for heat exchangers/heat sinks and reviewed for problems or errors; (2) the licensee utilized the periodic maintenance method outlined in EPRI NP-7552, "Heat Exchanger Performance Monitoring Guidelines;" (3) the licensee properly utilized biofouling controls; (4) the licensee's heat exchanger inspections adequately assessed the state of cleanliness of their tubes, and (5) the heat exchanger was correctly categorized under the Maintenance Rule.

Documents reviewed by the inspectors included:

- EPRI Report "Alternative to Thermal Performance Testing and/or Tube-Side Inspections of Air-to-Water Heat Exchangers," October 2002, T. Eckert
- Commitment Change Evaluation CCE-2007-0001
- Engineering Standard EN-EP-S-039-G, "Testing Standard for Safety-Related Heat Exchangers Cooled by Standby Service Water," Revision 0
- WO 97567

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11)

a. Inspection Scope

The inspectors observed testing and training of senior reactor operators and reactor operators to assess training, operator performance, and the evaluator's critique. The training scenario, GSMS-LOR-ONX02, Revision 05, involved high vibration on the reactor feed pump and a subsequent pump trip followed by a reactor recirculation flow

control valve runback. The scenario then simulated two control rod drifts which required the operators to manually scram the reactor in accordance with the off-normal event procedure for control rod drive malfunction. Documents reviewed by the inspectors are listed in the attachment.

Documents reviewed by the inspectors included:

- Procedure 01-S-04-2, "Licensed Operator Requalification Training," Revision 015
- GSMS-LOR-EXPEC, "Expectations of Performance during Simulator Training," Revision 07
- Training Procedure 14-S-02-20, "Preparing, Conducting and Review of Simulator Evaluations," Revision 004

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors reviewed the following three maintenance rule scoped systems that have displayed performance problems to: (1) verify the appropriate handling of structure, system, and component (SSC) performance or condition problems; (2) verify the appropriate handling of degraded SSC functional performance; (3) evaluate the role of work practices and common cause problems; and (4) evaluate the handling of SSC issues reviewed under the requirements of the maintenance rule, 10 CFR Part 50 Appendix B, and the TS.

- November 5, 2007, Remote shutdown panel system (C61)
- December 19, 2007, Reactor water cleanup system (G33/G36)
- December 20, 2007, Emergency switchgear and battery room ventilation (Z77)

Documents reviewed by the inspectors are included in the attachment.

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

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

.1 Risk Assessment and Management of Risk

a. Inspection Scope

The inspectors reviewed the listed assessment activities to verify: (1) performance of risk assessments when required by 10 CFR 50.65 (a)(4) and licensee procedures prior to changes in plant configuration for maintenance activities and plant operations; (2) the accuracy, adequacy, and completeness of the information considered in the risk assessment; (3) that the licensee recognized, and/or entered as applicable, the appropriate licensee-established risk category according to the risk assessment results and licensee procedures; and (4) that the licensee-identified and corrected problems related to maintenance risk assessments.

- WO117322, Plant Service Water Radial Well Pump D Maintenance
- Risk impact of thunderstorm warning on November 21, 2007

Documents reviewed by the inspectors are included in the attachment.

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

.2 Emergent Work Control

a. Inspection Scope

For the work activities listed below, the inspectors: (1) verified that the licensee performed actions to minimize the probability of initiating events and maintained the functional capability of mitigating systems and barrier integrity systems; (2) verified that emergent work-related activities such as troubleshooting, work planning/scheduling, establishing plant conditions, aligning equipment, tagging, temporary modifications, and equipment restoration did not place the plant in an unacceptable configuration; and (3) reviewed the UFSAR to determine if the licensee identified and corrected risk assessment and emergent work control problems.

- WO108076, ECCS Room Cooler Acid Flush
- WO95539, Reactor Heat Removal System Train C Breaker Replacement

Documents reviewed by the inspectors are included in the attachment.

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors: (1) reviewed plant status documents such as operator shift logs, emergent work documentation, deferred modifications and standing orders to determine if an operability evaluation was warranted for degraded components; (2) referred to the UFSAR and design basis documents to review the technical adequacy of licensee operability evaluations; (3) evaluated compensatory measures associated with operability evaluations; (4) determined degraded component impact on any TS; (5) used the Significance Determination Process to evaluate the risk significance of degraded or inoperable equipment; and (6) verified that the licensee has identified and implemented appropriate corrective actions associated with degraded components.

- CR-GGN-2007-04997, Standby Service Water Valve P41F241 Leaking
- CR-GGN-2007-05046, Division III Standby Diesel Generator Jacket Water Temperature Low
- CR-GGN-2007-04227, Standby Service Water Train C piping Corrosion and Repair
- CR-GGN-2007-04821, Division II Standby Diesel Generator Low Lube Oil Pressure Alarm
- CR-GGN-2007-05395, Electro-Hydraulic Pressure Control Causes Oscillations in Pressure and Power
- CR-GGN-2007-05488, Standby Service Water Pump House Ventilation Damper Failures

Documents reviewed by the inspectors are included in the attachment.

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors selected the three listed postmaintenance test activities of risk significant systems or components. For each item, the inspectors: (1) reviewed the applicable licensing basis and/or design-basis documents to determine the safety functions; (2) evaluated the safety functions that may have been affected by the maintenance activity; and (3) reviewed the test procedure to ensure it adequately tested the safety function that may have been affected. The inspectors either witnessed or reviewed test data to verify that acceptance criteria were met, plant impacts were evaluated, test equipment was calibrated, procedures were followed, jumpers were properly controlled, test data results were complete and accurate, test equipment was removed, the system was properly re-aligned, and deficiencies during testing were documented. The inspectors also reviewed the UFSAR to determine if the licensee identified and corrected problems related to postmaintenance testing.

- WO117322, Plant service water radial well Pump D outlet flow control valve replacement postmaintenance test
- WO129247, Standby diesel generator postmaintenance test following an extended diesel maintenance outage
- WO119453, Replacement and retest of plant air system compressor

Documents reviewed by the inspectors are included in the attachment.

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the UFSAR, procedure requirements, and TS to ensure that the three listed surveillance activities demonstrated that the SSCs tested were capable of performing their intended safety functions. The inspectors either witnessed or reviewed test data to verify that the following significant surveillance test attributes were adequate: (1) preconditioning; (2) evaluation of testing impact on the plant; (3) acceptance criteria; (4) test equipment; (5) procedures; (6) jumper/lifted lead controls; (7) test data; (8) testing frequency and method demonstrated TS operability; (9) test equipment removal; (10) restoration of plant systems; (11) fulfillment of ASME Code requirements; (12) updating of performance indicator (PI) data; (13) engineering

evaluations, root causes, and bases for returning tested SSCs not meeting the test acceptance criteria were correct; (14) reference setting data; and (15) annunciator and alarm setpoints. The inspectors also verified that the licensee identified and implemented any needed corrective actions associated with the surveillance testing.

- October 29, 2007, observed performance of the reactor core isolation cooling system surveillance performed per Procedure 06-OP-1E51-M-0001, "RCIC Monthly Run," Revision 3

- October 27, 2007, observed performance of the drywell purge compressor flow test performed per Procedure 06-OP-1E61-R-0011, "Drywell Purge Compressor Flow Verification," Revision 101

- November 11, 2007, reviewed inservice test of the reactor recirculation system valves per Procedure 06-OP-B33-Q-0003, "Reactor recirculation Quarterly Valve Test," Revision 102

- November 17, 2007, observed performance of control rod operability surveillance performed per Procedure 06-OP-1C11-M-0001, "Control Rod Operability," Revision 103

Documents reviewed by the inspectors are included in the attachment.

The inspectors completed four samples.

b. Findings

Introduction. The inspectors identified a noncited violation of Criterion V, "Instructions, Procedures, and Drawings," of 10 CFR Part 50, Appendix B, because the licensee failed to demonstrate compliance with Technical Specification Surveillance Requirement 3.5.3.1 due to an inadequate surveillance procedure.

Description. The inspector noted that the reactor core isolation cooling (RCIC) system is vented at the injection valve through a hard-piped drain with no visual means of detecting air in the system. Surveillance Requirement 3.5.3.1 requires that every 31 days the licensee must "Verify the RCIC System piping is filled with water from the pump discharge valve to the injection valve." To ensure that Surveillance Requirement 3.5.3.1 is met, the licensee implements Surveillance Procedure 06-OP-1E51-M-0001, "RCIC System Operability Verification." The procedure states to "Verify the discharge piping is full of water by opening the system high Points F205 and F206 and verify water is obtained per Data Sheet I." Data Sheet I only provides an initial block to show step completion and does not detail how to verify the discharge piping is full of water. The inspectors determined that the procedure failed to provide adequate acceptance criteria to qualitatively or quantitatively assess that the piping is full of water. On the basis of these inadequacies the inspectors determined that intent of the surveillance requirement, to verify the system is filled with water, could not be satisfied. This issue was entered into the licensee's CAP as Condition Report

CR-GGN-2007-03818. The licensee has added steps to the RCIC monthly and quarterly surveillance procedures to vent the system for a minimum of two minutes and has issued an engineering change request to modify system piping to add visual indication to system vent piping.

Analysis. The inspectors determined that the inadequate procedure was a performance deficiency because it failed to ensure the intent of Technical Specification Surveillance Requirement 3.5.3.1 was met. The finding was greater than minor because it affects the procedure quality attribute of the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using the Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 worksheet, the finding was determined to have a very low safety significance in that it did not result in the actual loss of the reactor core isolation cooling system, and was not potentially risk-significant due to external initiating events.

Enforcement. 10 CFR Part 50, Appendix B, Criterion V, requires that activities affecting quality shall be prescribed by documented procedures of a type appropriate to the circumstances and shall be accomplished in accordance with these procedures. The procedures shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to the above, the licensee failed to ensure that Surveillance Procedure 06-OP-1E51-M-0001 would verify that the reactor core isolation cooling system was full of water because the procedure failed to include appropriate quantitative or qualitative acceptance criteria for determining that venting of the system is satisfactorily accomplished. Because the violation was of very low safety significance and has been entered into the licensee's CAP as CR-GGN-2007-03818, this violation is being treated as a noncited violation, consistent with Section VI.A of the Enforcement Policy: NCV 05000416/2007005-01; Inadequate Procedure used to Vent the Reactor Core Isolation Cooling System.

1EP1 Exercise Evaluation (71114.01)

a. Inspection Scope

The inspectors reviewed the objectives and scenario for the 2007 biennial emergency plan exercise to determine if the exercise would acceptably test major elements of the emergency plan. The scenario simulated a release of carbon dioxide in the control building, the trip of a motor generator set, the failure of automatic and manual reactor scrams, fission product barrier failures, core damage and a radiological release to the environment via an unisolable steam line break in the turbine building to demonstrate the licensee's capabilities to implement the emergency plan.

The inspectors evaluated exercise performance by focusing on the risk-significant activities of event classification, offsite notification, recognition of offsite dose consequences, and development of protective action recommendations in the simulator control room and the following dedicated emergency response facilities:

- Technical Support Center
- Operations Support Center
- Emergency Operations Facility

The inspectors also assessed recognition of and response to abnormal and emergency plant conditions, the transfer of decision making authority and emergency function responsibilities between facilities, onsite and offsite communications, protection of emergency workers, emergency repair evaluation and capability and the overall implementation of the emergency plan to protect public health and safety and the environment. The inspectors reviewed the current revision of the facility emergency plan, and emergency plan implementing procedures associated with operation of the above facilities and performance of the associated emergency functions. These procedures are listed in the Attachment to this report.

The inspectors compared the observed exercise performance with the requirements in the facility emergency plan; 10 CFR 50.47(b); 10 CFR Part 50, Appendix E; and with the guidance in the emergency plan implementing procedures and other federal guidance.

The inspectors attended the post-exercise critiques in each of the above facilities to evaluate the initial licensee self-assessment of exercise performance. The inspectors also attended a subsequent formal presentation of critique items to plant management.

The inspectors completed one sample during the inspection.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

For the below listed simulator-based training evolution contributing to Drill/Exercise Performance and emergency response organization Performance Indicators, the inspectors: (1) observed the training evolution to assess classification, notification, and Protective Action Requirement development activities; (2) compared identified weaknesses and deficiencies against licensee identified findings to determine whether the licensee is properly identifying failures; and (3) determined whether licensee performance is in accordance with the guidance of the Nuclear Energy Institute (NEI) 99-02, "Voluntary Submission of Performance Indicator Data," acceptance criteria.

- November 5, 2007, observed a table top drill involving a tornado identified in the owner controlled area causing a loss of offsite power and subsequent station blackout with a loss of reactor core isolation cooling.

Documents reviewed by the inspectors included:

- Drill Emergency Notification Forms
- Procedure 10-S-01-1, “Activation of the Emergency Plan,” Revision 116
- EPP 01-02 Flow Chart

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control To Radiologically Significant Areas (71121.01)

a. Inspection Scope

This area was inspected to assess the licensee’s performance in implementing physical and administrative controls for airborne radioactivity areas, radiation areas, high radiation areas, and worker adherence to these controls. The inspector used the requirements in 10 CFR Part 20, the technical specifications, and the licensee’s procedures required by technical specifications as criteria for determining compliance. During the inspection, the inspector interviewed the radiation protection manager, radiation protection supervisors, and radiation workers. The inspector performed independent radiation dose rate measurements and reviewed the following items:

- Performance indicator events and associated documentation packages reported by the licensee in the Occupational Radiation Safety Cornerstone
- Adequacy of the licensee’s internal dose assessment for any actual internal exposure greater than 50 millirem committed effective dose equivalent
- Self-assessments, audits, licensee event reports, and special reports related to the access control program since the last inspection
- Corrective action documents related to access controls
- Changes in licensee procedural controls of high dose rate - high radiation areas and very high radiation areas
- Posting and locking of entrances to all accessible high dose rate - high radiation areas and very high radiation areas
- Radiation worker and radiation protection technician performance with respect to radiation protection work requirements

The inspector completed 11 of the required 21 samples.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

The inspector assessed licensee performance with respect to maintaining individual and collective radiation exposures as low as is reasonably achievable (ALARA). The inspector used the requirements in 10 CFR Part 20 and the licensee's procedures required by technical specifications as criteria for determining compliance. The inspector interviewed licensee personnel and reviewed:

- Current 3-year rolling average collective exposure
- Site-specific trends in collective exposures, plant historical data, and source-term measurements
- Site-specific ALARA procedures
- Three work activities of highest exposure significance completed during the last outage
- ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements
- Intended versus actual work activity doses and the reasons for any inconsistencies
- Integration of ALARA requirements into work procedure and radiation work permit (or radiation exposure permit) documents
- Person-hour estimates provided by maintenance planning and other groups to the radiation protection group with the actual work activity time requirements
- Post-job (work activity) reviews
- Method for adjusting exposure estimates, or re-planning work, when unexpected changes in scope or emergent work were encountered
- First-line job supervisors' contribution to ensuring work activities are conducted in a dose efficient manner
- Records detailing the historical trends and current status of tracked plant source terms and contingency plans for expected changes in the source term due to changes in plant fuel performance issues or changes in plant primary chemistry

- Source-term control strategy or justifications for not pursuing such exposure reduction initiatives

- Specific sources identified by the licensee for exposure reduction actions, priorities established for these actions, and results achieved since the last refueling cycle

- Self-assessments, audits, and special reports related to the ALARA program since the last inspection

- Resolution through the corrective action process of problems identified through post-job reviews and post-outage ALARA report critiques

- Corrective action documents related to the ALARA program and follow-up activities, such as initial problem identification, characterization, and tracking

- Effectiveness of self-assessment activities with respect to identifying and addressing repetitive deficiencies or significant individual deficiencies

The inspector completed 10 of the required 15 samples and 8 of the optional samples.

b. Findings

Introduction. The inspector reviewed a self-revealing, Green, noncited violation of TS 5.4.1.a for the failure to provide appropriate instructions in the work order package to properly isolate a differential pressure transmitter prior to opening the drain valve.

Description. On July 9, 2007, an instrumentation and control (I&C) maintenance worker received an electronic dosimeter alarm while draining condensation from a differential pressure transmitter in the Off-Gas System. This system contains primarily short-lived non-condensable noble gases.

Work order package 115745 was written to allow I&C technicians to isolate and drain condensation from pressure transmitter 1N64N006B, Preheater/Recombiner 'B' Differential Pressure Transmitter. Section 4.1 of the work order task instructions stated, "Close the isolation valves at the top of cabinet for the high side and low side sensing lines for Transmitter 1N64N006B." Section 4.2 stated, "Open the drain valves at the bottom of the cabinet for the high side and low side." When the I&C technicians opened the drain valves, they heard continuous gas flow from the transmitter. The radioactivity in the gas caused an electronic dosimeter dose rate alarm for one of the technicians. The drain valves were immediately shut. The dosimeter rate alarm stopped as soon as the gas dispersed. Both technicians exited the area and reported to radiation protection. Radiation protection immediately performed a survey of the affected area and determined that radiation levels were normal and there was no contamination of the work area. Both I&C technicians were able to pass through the contamination monitors at the exit of the radiologically controlled area without an alarm.

The licensee performed a review of the event and found the work order instructions did not completely isolate the transmitter from process flow. The wording in the work order was too generic with respect to isolation of the low and high point vents and did not identify that a total of four isolation valves were required to be closed in order to fully isolate the transmitter. In addition to the lack of specific valve information in the work order, some of the isolation valves had multiple labels. Corrective actions included modification of the work order to include all isolation valves and modification of the valve labels. Based on discussions with licensee personnel and the results of the investigation, the inspector concluded that the work order reviewer had not used appropriate references to provide instructions for safely isolating the transmitter.

Analysis. The failure to provide appropriate instructions in the work order package to properly isolate Transmitter IN64N006B is a performance deficiency which resulted in an electronic dosimeter alarm and the potential from unplanned, unintended radiation exposure for two technicians. The finding is more than minor because it is associated with the occupational radiation safety attribute of program and process and affected the cornerstone objective because it involves unplanned and unintended radiation exposure to workers. Using the Occupational Radiation Safety Significance Determination Process, the inspector determined that the finding was of very low safety significance because: it did not involve: (1) ALARA planning and controls, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. In addition, this finding has a crosscutting aspect in the area of work control associated with work planning because the licensee failed to properly plan work activities by incorporating specific plant system details into the work order to allow the I&C technicians to properly drain a pressure transmitter [H.3(a)].

Enforcement. TS 5.4.1.a, states that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A. Section 9a states that maintenance that can affect the performance of safety-related equipment should be properly preplanned and performed in accordance with written procedures, documented instructions, or drawings appropriate to the circumstances. Section 5.2[1]c of Procedure EN-WM-105, "Planning" states, in part, that "the licensee will perform task reviews by consulting planning references to get an understanding of the equipment, its function, and operational characteristics, including vendor manuals, plant drawings, and equipment history." Contrary to the above, on July 9, 2007**,** the licensee failed to perform an adequate task review for Work Order No.115745 in that they did not consult appropriate planning references to adequately isolate the pressure transmitter. Consequently, two technicians were exposed to unexpected radiation because they did not properly isolate the pressure transmitter prior to draining condensation from the system. Because the finding was of very low safety significance and was documented in the licensee's CAP as Condition Reports CR-GGN-2007-3461 and CR-GGN-2007-3477, this finding is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000416/2007005-02; Inadequate Procedure.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

Cornerstone: Mitigating Systems

The inspectors sampled licensee submittals for the performance indicator listed below for the period from January 2007 to December 2007. The definitions and guidance of NEI 99-02, “Regulatory Assessment Indicator Guideline,” Revision 2, were used to verify the licensee’s basis for reporting each data element in order to verify the accuracy of PI data reported during the assessment period. The inspectors reviewed licensee event reports, out-of-service logs, operating logs, and the maintenance rule database as part of the assessment.

- Mitigating Systems Performance Indicator
  -high pressure coolant injection systems
  -heat removal systems
  -emergency AC power system
  -residual heat removal system
  -supporting cooling water systems

The inspector completed five samples during the inspection.

Cornerstone: Occupational Radiation Safety

Occupational Exposure Control Effectiveness

The inspector reviewed licensee documents from April 1 through September 30, 2007. The review included corrective action documentation that identified occurrences in locked high radiation areas (as defined in the licensee’s technical specifications), very high radiation areas (as defined in 10 CFR 20.1003), and unplanned personnel exposures (as defined in NEI 99-02, "Regulatory Assessment Indicator Guideline," Revision 5). Additional records reviewed included ALARA records and whole body counts of selected individual exposures. The inspector interviewed licensee personnel that were accountable for collecting and evaluating the performance indicator data. In addition, the inspector toured plant areas to verify that high radiation, locked high radiation, and very high radiation areas were properly controlled. Performance indicator definitions and guidance contained in NEI 99-02, Revision 5, were used to verify the basis in reporting for each data element.

The inspector completed the required sample (1) in this cornerstone.

Cornerstone: Public Radiation Safety

Radiological Effluent Technical Specification/Offsite Dose Calculation Manual Radiological Effluent Occurrences

The inspector reviewed licensee documents from April 1 through September 30, 2007. Licensee records reviewed included corrective action documentation that identified occurrences for liquid or gaseous effluent releases that exceeded performance indicator thresholds and those reported to the NRC. The inspector interviewed licensee personnel that were accountable for collecting and evaluating the performance indicator data. Performance indicator definitions and guidance contained in NEI 99-02, Revision 5, were used to verify the basis in reporting for each data element.

The inspector completed the required sample (1) in this cornerstone.

Cornerstone: Emergency Preparedness

The inspector reviewed licensee evaluations for the three emergency preparedness cornerstone performance indicators of Drill and Exercise Performance, Emergency Response Organization Participation, and Alert and Notification System Reliability, for the period April 2006 through September 2007. The definitions and guidance of Nuclear Energy Institute Report 99-02, "Regulatory Assessment Indicator Guideline," Revisions 3 through 5, and the licensee Performance Indicator Procedures EN-LI-201, "Performance Indicators," Revision 6; 10-S-04-4, "Performance Indicators," Revision 6; and 01-S-10-3, "Emergency Preparedness Department Responsibilities," Revision 14, were used to verify the accuracy of the licensee's evaluations for each performance indicator reported during the assessment period.

The inspector reviewed a sample of drill and exercise scenarios and licensed operator simulator training sessions, notification forms, and attendance and critique records associated with training sessions, drills, and exercises conducted during the verification period. The inspector reviewed 20 selected emergency responder qualification, training, and drill participation records. The inspector reviewed alert and notification system testing procedures, maintenance records, and a one hundred percent sample of siren test records.

The inspector completed three samples during the inspection.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a daily screening of items entered into the licensee's CAP. This assessment was accomplished by reviewing work orders and CRs and attending corrective action review and work control meetings. The inspectors: (1) verified that equipment, human performance, and program issues were being identified by the licensee at an appropriate threshold and that the issues were entered into the CAP; (2) verified that corrective actions were commensurate with the significance of the issue; and (3) identified conditions that might warrant additional follow-up through other baseline inspection procedures.

The inspectors reviewed the exercise scenarios for three previous exercises, and reviewed drill evaluation reports for the period January 2006 through November 2007, to identify trends in emergency response organization performance, and reviewed exercise performance against identified trends.

The inspectors also evaluated the effectiveness of the licensee's problem identification and resolution process with respect to the following inspection areas:

- Access Control to Radiologically Significant Areas (Section 2OS1)
- ALARA Planning and Controls (Section 2OS2)

b. Findings

No findings of significance were identified.

.2 Selected Issue Follow-up Inspection

a. Inspection Scope

In addition to the routine review, the inspectors selected the two listed issues for a more in-depth review. The inspectors considered the following during the review of the licensee's actions: (1) complete and accurate identification of the problem in a timely manner; (2) evaluation and disposition of operability/reportability issues; (3) consideration of extent of condition, generic implications, common cause, and previous occurrences; (4) classification and prioritization of the resolution of the problem; (5) identification of root and contributing causes of the problem; (6) identification of corrective actions; and (7) completion of corrective actions in a timely manner.

- CR-GGN-2007-04227, Pitting Identified in Reactor Heat Removal Train C Piping

- CR-GGN-2007-04626, Main Turbine Seal Steam Controller Troubleshooting and Coincident Seal Steam Bypass Valve Failure Causes a Loss of Condenser Vacuum

- Operator Workarounds

Documents reviewed by the inspectors are listed in the attachment.

b. Findings and Observations

Introduction: The inspectors identified a Green finding involving a loss of condenser vacuum caused by improper troubleshooting of the seal steam pressure controller.

Description: On September 18, 2007, maintenance technicians were troubleshooting a seal steam pressure regulating valve and controller due to erratic valve movements. During troubleshooting activities the seal steam bypass valve was closed per the system operating instructions. The technicians inserted pressure input step changes into the controller circuitry of the regulating valve to verify the controller response. After several open and close deviations, the seal steam pressure controller became erratic and seal steam header pressure began oscillating from zero inches of water column to greater than 40 inches of water column. The control room returned the regulating valve to manual control at the moment the valve had oscillated to the closed position, thus removing all steam supply to the steam seal header. As a result, main condenser vacuum rapidly lowered requiring control room operators to reduce reactor power from 80 percent to 74 percent. Operators attempted to open the bypass valve from the control room, however the valve failed to fully open to restore seal steam pressure. Operators then attempted to open the seal steam control valve in manual mode, which opened the valve, restoring the seal steam header pressure and stabilized main condenser vacuum. During the event, high pressure condenser vacuum dropped from 27.2 to 23.5 inches of mercury. Operators were able to recover from the low vacuum condition prior to an automatic turbine trip set at 21 inches of mercury.

The inspectors reviewed the maintenance instructions used for troubleshooting the seal steam pressure controller. The inspectors determined that the troubleshooting instructions were inadequate. Specifically, the instructions simply stated, “Troubleshoot/ monitor/investigate identified problem to determine cause of valve failure,” and failed to ensure that during the troubleshooting evolution adequate seal steam pressure would be maintained. In addition, the inspectors determined that the licensee had failed to use Entergy corporate Procedure EN-MA-125, “Troubleshooting Control of Maintenance Activities.” EN-MA-125 requires that a risk evaluation is performed for the troubleshooting activity and requires management review and approval prior to implementation.

Analysis: The performance deficiency involved the failure to provide adequate work instructions for the troubleshooting of the seal steam pressure controller. The finding was more than minor because it was associated with the initiating events cornerstone attribute of equipment performance and affected the associated cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during power operations. Using the MC 0609, "Significance Determination Process," Phase 1 Worksheet, the inspectors concluded that a Phase 2 evaluation was required because the finding impacted both the initiating event and mitigating systems cornerstone. The inspectors performed a Phase 2 analysis using Appendix A,

"Determining the Significance of Reactor Inspection Findings for At-Power Situations," of Manual Chapter 0609, "Significance Determination Process," and the Phase 2 Worksheets for Grand Gulf Nuclear Station. The inspectors determined there was an increase in likelihood of a transient without the power conversion system but there was no reduction in remaining capability. Because the exposure time of the finding was more than 3 days, the results of the Phase 2 analysis were that the finding had very low safety significance.

The cause of the finding was related to the human performance crosscutting component of decision making, in that the licensee failed to use conservative assumptions during troubleshooting activities, performing these activities without evaluating the risk involved in troubleshooting the steam seal system and failed to identifying possible unintended consequences by not performing a required risk evaluation [H.1(b)]. This issue was entered into the licensee's CAP as condition Report CR-GGN-2007-04626.

Enforcement: No violation of NRC requirements occurred. FIN 0500416/2007005-03; Improper Control of Troubleshooting Causes a Loss of Condenser Vacuum.

.3 Semiannual Trend Review

a. Inspection Scope

The inspectors completed a semiannual trend review of repetitive or closely related issues that were documented in CRs, maintenance work orders, system health reports, and corrective action trend reports to identify trends that might indicate the existence of more safety significant issues. The inspectors review consisted of the six month period from June 30 through December 31, 2007. When warranted, some of the samples expanded beyond those dates to fully assess the issue. The inspectors reviewed the following issues:

- Loose item control and plant housekeeping
- Control rod operability and reactivity management
- Apparent cause documentation quality and issue resolution
- Foreign material control

The inspectors compared and contrasted their results with the results contained in the licensee's quarterly trend reports. Corrective actions associated with a sample of the issues identified in the licensee's trend report were reviewed for adequacy. Documents reviewed by the inspectors are listed in the attachment.

b. Findings and Observations

No findings of significance were identified. However, during the review the inspectors noted the following trends where performance deficiencies have recurred:

- The protective cover was found unattached on the reactor heat removal Train A heat exchanger outlet temperature detector. This was previously identified by the inspectors.

- Several NRC and licensee identified performance deficiencies have been identified during this six month period regarding loose item control and housekeeping issues.

4OA3 Event Follow-up (71153)

.1 Personnel Performance During Nonroutine Evolutions, Events, and Transients

a. Inspection Scope

The inspectors: (1) reviewed operator logs, plant computer data, and/or strip charts for the below listed evolutions to evaluate operator performance in coping with nonroutine events and transients; (2) verified that operator actions were in accordance with the response required by plant procedures and training; and (3) verified that the licensee has identified and implemented appropriate corrective actions associated with personnel performance problems that occurred during the events sampled.

- On October 19, 2007, the inspectors observed operator performance in the field and in the control room during a rapid power reduction due to a ruptured plant service water radial well pipe flange.

- On December 4, 2007, the inspectors responded to a loss of site communications after a communications cable had been severed by the local telephone company.

- On December 6, 2007, the inspectors observed operator performance in the control room during a rapid power reduction due to a loss of condenser vacuum event.

b. Findings

.1 Inadequate Engineering Review of Plant Service Water Modification

Introduction: A Green self-revealing finding was identified involving the failure to develop and implement an adequate modification for replacement of plant service water (PSW) flow control valves resulting in a circumferential pipe break.

Description: On October 19, 2007, a circumferential piping failure occurred on the flange of a cast iron pipe tee causing the PSW header pressure to rapidly decrease. Plant operators were alerted to the decreasing PSW header pressure by annunciation in the control room and responded by entering the off-normal event procedure for a partial loss of plant service water. Reactor power was reduced to approximately 45 percent in order to reduce the heat load on the PSW system. Locally, operators isolated the piping failure by manually closing an isolation valve resulting in restoration of PSW pressure.

The licensee determined that the cause of the pipe failure resulted from an inadequate modification of PSW system flow control valves. These valves were attached by a flanged connection to the tee piping that failed. The replacement valve had a raised face flange design that was not recognized in the engineering change review. In addition, the

package failed to provide proper assembly instructions of the valve to the tee flanged connection. Specifically, the proper bolting and gasket materials were not specified, nor were the bolting torque requirements included. These failures created excessive circumferential pipe stresses resulting in the pipe failure. The valve design modification was completed by contract engineering personnel. The licensee concluded that the engineers responsible for oversight of the modification did not properly review design requirements for the replacement valves and failed to properly review the modification and installation instructions.

Analysis: The performance deficiency involved the failure to develop and implement an adequate modification for replacement of PSW flow control valves. The finding was more than minor because it was associated with the initiating events cornerstone attribute of equipment performance and affected the associated cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during power operations. Using the MC 0609, "Significance Determination Process," Phase 1 worksheet, the finding was determined to have very low safety significance because the finding did not contribute to the likelihood that mitigating equipment or functions would not be available following a reactor trip. The cause of the finding was related to the human performance crosscutting component of work practices in that the responsible engineers failed to perform adequate self and peer checking during the development and review of the design modification to the PSW flow control check valves [H.4(a)]. This issue was entered into the licensee's CAP as Condition Report CR-GGN-2007-05040.

Enforcement: No violation of NRC requirements occurred. FIN 0500416/2007005-04; Inadequate Engineering Review of Plant Service Water Modification.

.2 Failure to Follow Procedure Results in Loss of Condenser Vacuum

Introduction: A Green self-revealing finding was identified involving a loss of condenser vacuum caused by plant operators improperly removing a steam jet air ejector (SJAE) from service.

Description: On December 6, 2007, the plant was operating with both SJAE trains in service and the control room was preparing to secure SJAE Train A. During the system rotation, step 5.1.2(hh) of System Operating Instruction (SOI) 04-1-01-N62-1, "Condenser Air Removal", required that the steam pressure controller for the jet being removed from service be set to approximately 50 psig. The step also states that the reduction in controller pressure is done to prevent large pressure oscillations once the SJAE steam isolation valve is closed. The control room supervisor (CRS) made the decision to lower the pressure controller to a zero psig setpoint versus the 50 psig as required per the step. The control room supervisor assumed that if the controller was set to a lower pressure control, steam pressure oscillations would be reduced. This decision to deviate from the SOI was not discussed with the shift manager, shift supervisor or reactor operators. Per the SOI, once the SJAE is removed from service by closing the steam isolation valves, then the operators are instructed to lower the steam controller setpoint to zero. In this case, the steam isolation valves were not closed prior to lowering the steam controller to zero psig, hence the intent of the system operating instructions

was compromised. When the setpoint was reduced to zero psig, a reverse flow condition occurred in the SJAE Train A, causing a loss of condenser vacuum. The off-normal event procedure for loss of condenser vacuum was entered and reactor power reduced to 76 percent of rated core thermal power. The steam controller for the SJAE Train A was restored to a setting of 50 psig, and condenser vacuum stabilized. The offgas reverse flow occurred for approximately seven minutes during which main condenser vacuum trended down from 27.6 to 24.5 inHg before recovering.

The licensee identified the root cause as a failure of plant management to recognize and correct procedural noncompliance. This resulted in the mindset of acceptance to deviate from procedural instructions and allowed the CRS to rationalize that there was an allowance to deviate from the procedure. This caused the SJAE rotation to change from being performed by an approved plant procedure to being performed in a knowledge space environment. Administrative procedural guidance, at the time of the event, did allow performing steps within an SOI out of sequence if the intent of the procedure was not compromised. However, performance of the SOI steps out of sequence did violate the intent.

Analysis: The performance deficiency involved the failure to remove a SJAE from service per the system operating instructions. The finding was more than minor because it was associated with the initiating events cornerstone attribute of equipment performance and affected the associated cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during power operations. Using the MC 0609, "Significance Determination Process," Phase 1 Worksheet, the inspectors concluded that a Phase 2 evaluation was required because the finding impacted both the initiating event and mitigating systems cornerstone. The inspectors performed a Phase 2 analysis using Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," of Manual Chapter 0609, "Significance Determination Process," and the Phase 2 Worksheets for Grand Gulf Nuclear Station. The inspectors determined there was an increase in likelihood of a transient without the power conversion system but there was no reduction in remaining capability. Because the exposure time of the finding was more than 3 days, the results of the Phase 2 analysis were that the finding had very low safety significance.

The cause of the finding was related to the human performance crosscutting component of work practices in that plant management failed to ensure supervisory and management oversight of work activities such that nuclear safety is supported [H.4(c)]. The licensee entered this issue into their CAP as Condition Report CR-GGN-2007-05676.

Enforcement: No violation of NRC requirements occurred; FIN 0500416/2007005-05, Failure to Follow Procedure Results in Loss of Condenser Vacuum.

4OA6 Meetings, Including Exit

On November 29, 2007, the inspector presented the occupational radiation safety inspection results to Mr. M. Krupa and other members of his staff who acknowledged the findings. The inspector confirmed that proprietary information was not provided or

examined during the inspection.

On December 13, 2007, the inspectors presented the inspection results to Mr. R. Brian, Site Vice President, and other members of his staff, who acknowledged the findings. The inspectors confirmed that proprietary information examined during the inspection had been returned to the designated custodians.

On January 7, 2008 the resident inspectors presented the inspection results to Mr. R. Brian and others who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

4OA7 Licensee-Identified Violations

The following violation of very low significance (Green) was identified by the licensee and is a violation of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as a NCV.

- TS 5.4.1(a) requires written procedures to be implemented as recommended by Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Appendix A recommends procedures for performing maintenance that can affect the performance of safety-related equipment. Contrary to this requirement, the maintenance procedure for replacement of the safety-related Riley temperature switch used for operation of the standby service water pump house ventilation recirculation damper did not have adequate instructions for proper instrument calibration. This resulted in the Riley temperature switch operating in reverse of the design. The recirculation damper is used to regulate temperatures in the standby service water pump house by allowing pump house air to be recirculated to the room in cold weather and no recirculating air in warm weather. This issue was documented in Condition Report CR-GGN-2007-05488. This finding is of very low safety significance since there was no actual loss of operability.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

D. Barfield, Director, Engineering
B. Blanche, Assistant Manager, Operations
C. Bottemiller, Manager, Plant Licensing
R. Brian, Vice President, Operations
M. Causey, Senior Lead Technical Specialist
R. Collins, Manager, Corrective Actions and Assessments
D. Cotton, Technician, Radiation Protection
D. Coulter, Licensing Specialist, Plant Licensing
D. Cupstid, Superintendent, Nuclear and Industrial Safety
P. Different, Senior Lead Engineer, Reactor Engineering
B. Edwards, Minority Owner Specialist
C. Ellsaesser, Manager, Operations
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S. Hedger, Operations Engineer
W. Walker, Senior Project Engineer, Reactor Project Branch C

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000416/2007005-01	NCV	Inadequate Procedure Used to Vent the Reactor Core Isolation Cooling System (Section 1R22)
05000416/2007005-02	NCV	Inadequate Procedure (Section 2OS2.b)
05000416/2007005-03	FIN	Improper Control of Troubleshooting Causes a Loss of Condenser Vacuum (Section 4OA2)
05000416/2007005-04	FIN	Inadequate Engineering Review of Plant Service Water Modification (Section 4OA3)
05000416/2007005-05	FIN	Failure to Follow Procedure Results in Loss of Condenser Vacuum (Section 4OA3)

Discussed

None

LIST OF DOCUMENTS REVIEWED

In addition to the documents called out in the inspection report, the following documents were selected and reviewed by the inspectors to accomplish the objectives and scope of the inspection and to support any findings:

Section 1R04: Equipment Alignment

Procedure 04-1-01-E12-1, “System Operating Instruction - Residual Heat Removal System,” Revision 128

Procedure 04-1-01-C41-1, “System Operating Instruction - Standby Liquid Control System,” Revision 116

Procedure 04-1-01-P81-1, “System Operating Instruction - High Pressure Core Spray Diesel Generator,” Revision 059

Procedure 04-1-01-P81-1, “System Operating Instruction - Fuel Pool Cooling and Cleanup System,” Revision 054

System Piping Diagram M-1085, “Residual Heat Removal System”
System Piping Diagram M-1082, “Standby Liquid Control System”
System Piping Diagram M-1093, “High Pressure Core Spray Diesel Generator”
System Piping Diagram M-1106, “High Pressure Core Spray Diesel Generator”
System Piping Diagram M-1088, “Fuel Pool Cooling and Cleanup System”

Vendor Manual 460000154
Vendor Manual 460000155
Vendor Manual 460000962
GEK-73689, “Fuel Pool Cooling and Cleanup System”

Section 1R06: Flood Protection Measures

Off-Normal Event Procedure 05-1-02-VI-1, “Flooding,” Revision 105
GGNS Safety Evaluation Report Section 10.4.5
Regulatory Guide 1.102
CR-GGN-2007-04496
CR-GGN-2007-05522
CR-GGN-2007-05683
WO 117324

Section 1R12: Maintenance Rule

EN-DC-203, “Maintenance Rule Program,” Revision 0
EN-DC-204, “Maintenance Rule Scope and Basis,” Revision 0
EN-DC-205, “Maintenance Rule Monitoring,” Revision 0
EN-DC-206, “Maintenance Rule (a)(1) Process,” Revision 0
Maintenance Rule Failure Database for System Z77
Maintenance Rule Failure Database for System G33/G36
Maintenance Rule Failure Database for System C61
GGNS Maintenance Rule (a)(1) systems Report, October 2007
GGNS Maintenance Rule (a)(1) systems Report, December 2007
Maintenance Rule System Notebook

Condition Reports for Z77 system
CR-GGN-2007-05370
CR-GGN-2007-04977
CR-GGN-2007-04959
CR-GGN-2007-03760
CR-GGN-2007-03055

Condition Reports for G33/G36 system
CR-GGN-2007-05898
CR-GGN-2007-05430
CR-GGN-2007-05107

Condition Reports for C61 system
CR-GGN-2007-02583
CR-GGN-2007-02265
CR-GGN-2007-01915
CR-GGN-2007-01585
CR-GGN-2007-01403

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

01-S-18-6, "Risk Assessment of Maintenance Activities," Revision 5
18-S-01-1, "Planning Guidelines," Revision 2
EN-WM-101, "On-Line Work Management Process," Revision 1
EN-WM-102, "Work Implementation and Closeout," Revision 0

WO 117322
WO 95539
WO 108076

Section 1R15: Operability Determinations

EN-OP-104, "Operability Determinations," Revision 2
EN-LI-102, "Corrective Action Process," Revision 12

CR-GGN-2007-04997
CR-GGN-2007-05046
CR-GGN-2007-04227
CR-GGN-2007-04821
CR-GGN-2007-05395
CR-GGN-2007-05488

Section 1R19: Postmaintenance Testing

EN-MA-101, "Conduct of Maintenance," Revision 5
EN-MA-114, "Post Maintenance Testing," Revision 5

WO117322
WO129247
WO119453
CR-GGN-2007-05819
CR-GGN-2007-05760
CR-GGN-2007-05212
CR-GGN-2007-05281

Section 1R22: Surveillance Testing

Procedure 01-S-06-12, "GGNS Surveillance Program," Revision 109
Procedure 06-OP-1E51-M-0001, "RCIC Monthly Run," Revision 3
Procedure 06-OP-1E61-R-0011, "Drywell Purge Compressor Flow Verification," Revision 101
Procedure 06-B33-Q-0003, "Reactor Recirculation Quarterly Valve Test," Revision 102
Procedure 06-OP-1C11-M-0001, "Control Rod Operability," Revision 103
Procedure EN-OP-115, "Conduct of Operations," Revision 5
CR-GGN-2007-3818
LO-NOE-2006-00400

Section 1EP1: Exercise Evaluation

10-S-01-01, "Activation of the Emergency Plan," Revision 116

10-S-01-12, "Radiological Assessment and Protective Action Recommendations," Revision 36

10-S-01-19, "Personnel Injury," Revision 14

10-S-01-29, "Operations Support Center Operations," Revision 20

10-S-01-30, "Technical Support Center Operations," Revision 13

10-S-01-33, "Emergency Operations Facility Operations," Revision 15

10-S-01-06, "Notification of Offsite Agencies and Plant On-Call Emergency Personnel," Revision 44

2006-00429, “Annual Site Medical Drill”
2007-00361, “Annual Site Medical Drill”

Grand Gulf Emergency Plan, Revision 44

Section 2OS1: Access Controls to Radiologically Significant Areas

Condition Reports:
CR-GGN-2007-01834
CR-GGN-2007-01945
CR-GGN-2007-01952
CR-GGN-2007-02185
CR-GGN-2007-02241
CR-GGN-2007-02294
CR-GGN-2007-03519
CR-GGN-2007-03657
CR-GGN-2007-04203
CR-GGN-2007-05126
CR-GGN-2007-05569

Audits and Self-Assessments:
RF15 RP Outage Critique
QA-14-2007-GGNS-1, Quality Assurance Audit Report, Radiation Protection
LO-LGO-2007-0044, Access Control to Radiologically Significant Areas and PI Verification
GLO 2007-0146, Access Control to Radiologically Significant Areas and PI Verification

Procedures:
01-S-08-1 Administration of the GGNS Radiation Protection Program, Revision 104
01-S-08-2 Exposure and Contamination Control, Revision 117
08-S-02-50 Radiological Surveys and Surveillances, Revision 115
EN-RP-100 Radworker Expectations, Revision 0
EN-RP-101 Access Control for Radiologically Controlled Areas, Revision 2
EN-RP-108 Radiation Protection Posting, Revision 5

Section 2OS2: ALARA Planning and Controls

Condition Reports:
CR-GGN-2007-02045
CR-GGN-2007-02047
CR-GGN-2007-02763
CR-GGN-2007-02834
CR-GGN-2007-03461
CR-GGN-2007-03736
CR-GGN-2007-03477

Audits and Self-Assessments:
GLO-2007-0145, Program Control Self-Assessment
GLO-2007-0045, Program Control Self-Assessment

Radiation Work Permits:
2007-1310 'B' Recirculation Pump Motor Replacement
2007-1505 RF15 Scaffolding
2007-1508 RF15 Under Vessel Work Activities

Procedures:
01-S-07-1 Control of Work on Plant Equipment and Facilities, Revision 39
EN-RP-105 Radiation Work Permits, Revision 2
EN-RP-203 Dose Assessment, Revision 1
EN–WM-105 Planning, Revision 2

Miscellaneous Documents:
Work Order Package 00115745
1N64N006B Transmitter Work Instructions
Five Year ALARA Plan 2007 - 2011
ALARA Managers Committee Meeting Minutes; 8/31/07, 9/09/07, 11/06/07, 11/19/07
ALARA Sub-Committee Meeting Minutes; 11/06/07, 11/08/07, 11/16/07

Section 4OA2: Identification and Resolution of Problems

Lesson Plan: "Operation, Management and Maintenance of the Public ANS Sirens"
Standard GGNS-E-100.0
Environmental Qualification Document Package EQ08.1
Specification GE PPD 145C3224/117C3485, "Pyco Temperature Element"
CR-GGN-2007-04550

CR-GGN-2007-04626
CR-GGN-2007-04713
Operations Decision Making Instruction, “Seal Steam Pressure Control 1N33F505A/B & 1N33F105,” Revision 00
EN-MA-125, “Troubleshooting Control of Maintenance Activities,” Revision 3
WO-122959

Section 4OA3: Event Follow-up

Drawing M185.0-NSP47F002A-1.2-001 Revision A
GFIG-OPS-P4447, “Plant Service Water System, Figure 1"
GFIG-OPS-P4447, “Plant Service Water System, Figure 2"
CR-GGN-2004-1851
CR-GGN-2004-2933
CR-GGN-2007-2065
CR-GGN-2007-5040
WO117322
WO117325
ER-GG-1996-0030
ER-GG-1998-0070
ER-GG-2006-0030
General Maintenance Instruction 07-S-14-281, “Flanged Connection Bolting Procedure”
System Operating Instruction 04-1-01-N62-1, “Condenser Air Removal,” Revision 065
Drawing M-1060A, “Condenser Air Removal System,” Revision 31

Section 4OA7: Licensee-Identified Violations

Calculation XC-Q1Y41-92007, “SSW Pump house room temperatures for a LOCA and for SSW pump house cooling inoperable,” Revision 0

GGNS-92-0002, “The evaluation of safety related electrical equipment in various rooms with elevated post LOCA temperatures,” Revision 0,1,2

System Design Criteria Y47, “Standby Service Water Pump House Ventilation System,” Revision 0

Maintenance Calibration Instruction 07-S-53-169, “Riley Model 86 Thermocouple Monitor,” Revision 11

CR-GGN-2007-5488
MAI-294860
WO131212