UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
612 EAST LAMAR BLVD, SUITE 400
ARLINGTON, TEXAS 76011-4125

February 9, 2009

James R. Douet, 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 05000416/2008005

Dear Mr. Douet:

On December 31, 2008, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Grand Gulf Nuclear Station. The enclosed integrated inspection report documents the inspection findings, which were discussed on January 13, 2009, with Jeremy G. Browning, General Manager, Plant Operations and other members of your staff.

The inspections 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 nine NRC identified and self-revealing findings of very low safety significance (Green). Five of these findings were determined to involve violations of NRC requirements. Additionally, a licensee-identified violation, which was determined to be of very low safety significance, is listed in this report. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these findings as a noncited violations, consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest the violations or the significance of the 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, D.C. 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 612 E. Lamar Blvd, Suite 400, Arlington, Texas, 76011-4125; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 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, and its enclosure, will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records 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/

Geoffrey B. Miller, Chief
Project Branch C
Division of Reactor Projects

Docket: 50-416
License: NPF-29

Enclosure:
NRC Inspection Report 05000416/2008005
w/Attachment: Supplemental Information

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Publicly Avail	☒ Yes ☐ No	Sensitive	☐ Yes ☒ No	Sens. Type Initials	WCW

RIV:RI:DRP/C	SRI:DRP/C	C:SPE:DRP/C	C:DRS/EB1	C:DRS/PSB1
AJBarrett	RLSmith	WCWalker	TFarnholtz	MPShannon
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01/26/2009	01/26/2009	01/20/2009	01/22/2009	01/26/2009
C:DRS/OB	C:DRS/EB2	C:DRS/PSB2	C:DRP/C	
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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	50-416
License:	NPF-29
Report:	05000416/2008005
Licensee:	Entergy Operations, Inc.
Facility:	Grand Gulf Nuclear Station
Location:	Waterloo Road Port Gibson, MS
Dates:	September 22 through December 31, 2008
Inspectors:	R. Smith, Senior Resident Inspector A. Barrett, Resident Inspector D. Bollock, Project Engineer L. Carson II, Senior Health Physicist B. Henderson, Reactor Inspector S. Makor, Reactor Inspector P. Elkmann, Senior Emergency Preparedness Inspector
Approved By:	Geoffrey B. Miller, Chief, Project Branch C Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000416/2008005; 09/22/2008 – 12/31/2008; Grand Gulf Nuclear Station, Integrated Resident and Regional Report; Fire Protection, Maintenance Effectiveness, Operability Evaluations, Refueling and Other Outage Activities and Event Follow-up.

The report covered a 3-month period of inspection by resident inspectors and announced baseline inspections by regional based inspectors. Nine Green findings were identified by the inspectors. Six of these findings were considered noncited violations of NRC regulations. 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 review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 4, dated December 2006.

A. NRC-Identified Findings and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. The inspectors reviewed a self-revealing Green finding involving a recirculation pump trip during pump up-shift to fast speed due to ineffective corrective actions. The plant had recently replaced the recirculation motor on Pump A during the refuelling outage and during investigation determined that the instantaneous over-current trip for the breaker had drifted low. The inspectors performed a review of condition reports and determined that reactor recirculation Pump B had tripped following motor replacement for the same reason in September 2007. The licensee entered this issue into the corrective action program as Condition Report CR-GGN-2008-06269.

  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 Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the finding was determined to be of very low safety significance (Green) since it did not contribute to loss of function of mitigating equipment. The cause of this finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program in that the licensee failed to perform a thorough evaluation of a problem that resulted in a plant transient such that the resolution properly addressed the cause and extent of condition [P.1(c)]. (Section 1R20)

- Green. The inspectors reviewed a self-revealing Green finding involving an automatic reactor scram caused by an operator inadvertently closing steam supply valves to the reactor feed pump turbine. Site personnel investigating the scram determined that an operator had incorrectly performed actions for the reactor feed Pump B turbine on the reactor feed Pump A turbine control switches at a local panel. The operator inadvertently closed the steam supply valves to the reactor feed Pump A turbine resulting in a total loss of feedwater flow and low reactor water level scram. The licensee entered this issue into the corrective action program as Condition Report CR-GGN-2008-06195.

The finding was more than minor because it was associated with the initiating events cornerstone attribute of human 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 Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the inspectors concluded that an evaluation was required by the regional senior reactor analyst, because the finding impacted both the initiating event and mitigating systems cornerstone. The senior reactor analyst performed a Phase 3 analysis and determined the issue was very low safety significance (Green). The cause of this finding has a crosscutting aspect in the area of human performance associated with work practices because the operator failed to use proper self-checking techniques while performing actions to place feed Pump B in the standby lineup [H.4(a)]. (Section 4OA3)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding for fire brigade performance deficiencies that were not identified by the licensee during a fire drill critique. The inspectors identified several deficiencies during the drill including issues relating to command and control, fire fighting strategy and use of fire fighting equipment. The inspectors provided feedback to plant personnel on the identified performance issues and the inadequate drill evaluation. The licensee entered this issue into the corrective action program as Condition Report CR-GGN-2008-06522.

  This finding was more than minor because it was associated with the protection against external factors attribute of the reactor safety mitigating systems cornerstone objective and adversely affected the cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. Manual Chapter 0609, Appendix F, “Fire Protection Significance Determination Process,” was used to analyze the finding since the inadequate critique had an adverse effect on fire brigade effectiveness, in relation to defense-in-depth strategies. Manual Chapter 0609, Appendix F states that findings associated with the onsite manual fire brigade are excluded. Therefore, in accordance with Manual Chapter 0609, the safety significance was determined by regional management review. Regional management concluded that the finding was of very low safety significance because it reflected fire brigade performance during a training drill, rather than during an actual fire. The cause of this finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program because the licensee failed to have a low enough threshold in identifying performance issues associated with a plant fire drill [P.1(a)]. (Section 1R05)

- Green. The inspectors identified a noncited violation of 10 CFR 50.65(a)(2) for the failure to adequately monitor the performance of the engineering safety features electrical switchgear and battery room ventilation system. The inspectors identified a condition report from March 2004 that had not been screened and evaluated in the maintenance rule database as a maintenance preventable functional failure. The condition report identified a room cooler that had tripped due to excessive current on the fan motor because an incorrectly sized sheave was installed during previous maintenance. The licensee entered this issue into the corrective action program as Condition Report CR-GGN-2008-02219.

The inspectors determined that this finding was more than minor since the engineering safety features electrical switchgear and battery room ventilation system was not placed in (a)(1) monitoring status in a timely manner. In addition, the finding was more than minor since violations of 10 CFR 50.65(a)(2) necessarily involve degraded system performance, which, if left uncorrected, could become a more significant safety concern. This finding has very low safety significance because the maintenance rule aspect of the finding did not lead to an actual loss of safety function of the system or cause a component to be inoperable, nor did it screen as potentially risk significant due to a seismic, flooding, or severe weather-initiating event. (Section 1R12)

- <u>Green</u>. The inspectors identified a Green noncited violation of 10 CFR Part 50 Appendix B, Criterion V involving two examples of a failure to follow procedures which resulted in inadequate operability evaluations. The first example involved an inadequate evaluation of foreign material in the condensate storage tank. The evaluation relied on an assumption that the high-pressure core spray and reactor core isolation cooling pumps would not be damaged by metal debris entrained in the pumps suction. The second example involved an inadequate evaluation of the structural integrity of the standby service water cooling towers that only considered the loss of structural support from a single beam. The licensee entered these issues into the corrective action program as Condition Reports CR-GGN-2008-05685 and CR-GGN-2008-06044.

  This finding is more than minor because the failure to perform adequate operability evaluations, if left uncorrected, could become a more significant safety concern. Using the Manual Chapter 0609, “Significance Determination Process,” Phase 1 Worksheet, this finding was of very low safety significance since it did not result in a loss of operability, nor did it screen as potentially risk significant due to a seismic, flooding, or severe weather-initiating event. The cause of this finding has a crosscutting aspect in the area of human performance associated with decision making because licensee personnel failed to use conservative assumptions and did not verify the validity of the underlying assumptions used in making safety-significant decisions [H.1(b)]. (Section 1R15)

- <u>Green</u>. The inspectors identified a Green noncited violation of 10 CFR Part 50 Appendix B, Criterion XVI, involving the failure to take timely corrective actions for corrosion on distribution beam structural support posts in the standby service water basin cooling towers. The licensee entered this issue into the corrective action program as Condition Report CR-GGN-2008-05434.

  The finding was more than minor because it was associated with the protection against external factors attribute of the reactor safety mitigating systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Manual Chapter 0609, “Significance Determination Process,” Phase 1 Worksheet, the finding was determined to be of very low safety significance since it did not represent an actual loss of safety function of the standby service water cooling towers, nor did it screen as potentially risk significant due to a seismic, flooding, or severe weather-initiating event. The cause of this finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program because licensee personnel failed to identify issues completely, accurately, and in a timely manner commensurate with their safety significance [P.1(a)]. (Section 4OA3)

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, involving a failure to take corrective actions to prevent recurrence of severe corrosion in piping hangers, piping supports, and piping in the standby service water basin cooling towers. Significant corrosion of the standby service water supports in October 2008 had been previously identified by plant personnel during a ten-year in-service inspection on October 3, 1993. At that time, plant personnel determined this to be a significant degraded condition of a safety related system, requiring replacement of the piping and associated supports. The licensee entered this issue into the corrective action program as Condition Report CR-GGN-2008-05434.

  This finding was more than minor because the corrosion represented a degrading condition that if left uncorrected could become more significant safety concern. The finding was also more than minor because it was associated with the protection against external factors attribute of the reactor safety mitigating systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using the Manual Chapter 0609, “Significance Determination Process,” Phase 1 Worksheet, this finding was of very low safety significance since it did not represent an actual loss of safety function, nor did it screen as potentially risk significant due to a seismic, flooding, or severe weather-initiating event. (Section 4OA3)

Cornerstone: Barrier Integrity

- Green. The inspectors identified a Green noncited violation of 10 CFR Part 50 Appendix B, Criterion XVI, involving the failure to correct leaking reactor water cleanup system primary containment isolation valves. During refuelling Outage 16, plant personnel were performing local leak rate testing of reactor water cleanup backwash containment penetration. Testing determined that these primary containment isolation valves exceeded the allowable leakage rate by greater than 10 times the leakage limits. The inspectors determined that for four consecutive operating cycles, the site had failed to take corrective actions to correct the excessive leakage through these valves. The licensee entered this issue into the corrective action program as Condition Report CR-GGN-2008-05139.

  The finding was more than minor because it was associated with systems, structures, and components and the reactor coolant system barrier performance attribute of the barrier integrity cornerstone and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers would protect the public from radionuclide releases caused by accident or events. Using Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the finding was determined to be of very low safety significance (Green) since it did not represent an actual open pathway in the physical integrity of the containment system. The cause of this finding has a crosscutting aspect in the area of human performance associated with resources in that the licensee failed to take actions to correct a long-standing equipment issue associated with excessive leakage from primary containment isolation valves [H.2(a)]. (Section 1R20)

- Green. The inspectors identified a Green finding involving the failure to demonstrate proper monitoring of plant parameters to control reactor coolant system cooldown rate to within expected management standards. The plant experienced a reactor scram from approximately 15 percent power during plant start-up from a refuelling outage due to a total loss of feedwater. Reactor pressure decreased at a faster rate than expected due to low decay heat levels and the injection of relatively cold condensate storage tank water to reactor vessel. The control room supervisor did not give a pressure band after pressure decreased below the low end of the emergency operating procedure band of 800 psig or assign a licensed operator to monitor reactor pressure during the event. The inspectors identified to the operators that the plant was approaching the procedural limit for cooldown rate; operators then closed the inboard main steam isolation valves to prevent exceeding the cooldown rate. The licensee entered this issue into the corrective action program as Condition Report CR-GGN-2008-06201.

  The finding is more than minor since it affects the human performance attribute of the barrier integrity cornerstone and affects the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Using the Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, inspectors determined that the finding has very low safety significance (Green) since it did not represent an actual degradation of the radiological barrier function of the reactor coolant system barrier. The cause of this finding has a crosscutting aspect in the area of human performance associated with decision making because control room supervision failed to maintain proper oversight to ensure reactor coolant cooldown rate was maintained within procedural limits [H.4(c)]. (Section 4OA3)

B. Licensee-Identified Violations

Violations of very low safety significance, which were identified by the licensee, have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensees corrective action program. These violations and corrective action tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Grand Gulf Nuclear Station began the inspection period coasting down in power to Refueling Outage 16. On September 21, 2008, the operators inserted a manual scram to begin the refueling outage. On October 22, 2008, the plant was brought to critical and began power ascension. On October 23, 2008, at approximately 15 percent power, the plant experienced an automatic reactor scram due to low reactor water level following the trip of reactor feedwater Pump A. On October 25, 2008, the plant began startup. On October 26, 2008, the plant experienced an automatic reactor scram from approximately 50 percent power due to a generator trip caused by a fault in the turbine generator thyristor voltage regulator. On October 27, 2008, the plant began startup and achieved 100 percent power on November 4, 2008. On November 17, 2008, operators reduced power to 41 percent power due to a fire on reactor feedwater Pump B. Following repairs to the feedwater pump, power was increased and the plant returned to full rated power on November 20, 2008. On November 28, 2008, the plant reduced power to 91 percent following a failure in a cooling bank of the main transformer. The plant returned to 100 percent on the same day. On November 30, 2008, the plant reduced power to 90 percent due to another failure in a separate cooling bank of the main transformer. The plant returned to 100 percent power on the same day. On December 13, 2008, the plant reduced power to approximately 90 percent to perform monthly control rod surveillances. On December 14, power was returned to 100 percent. The plant remained at or near full rated thermal power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity and Emergency Preparedness

1R04 Equipment Alignments (71111.04)

.1 Partial Walkdowns

a. Inspection Scope

The inspectors performed partial system walkdowns of the following risk-significant systems:

- September 22, 2008, the inspectors walked down the Division I standby diesel generator during a planned permanent modification of the Division II standby diesel generator

- December 5, 2008, the inspectors walked down the plant radial well system including all radial well pump houses following system maintenance

The inspectors selected these systems based on their risk significance relative to the reactor safety cornerstones at the time they were inspected. The inspectors attempted to identify any discrepancies that could affect the function of the system, and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, Updated Final Safety Analysis Report, technical specification requirements, administrative technical specifications, outstanding work orders, condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the corrective action program with the appropriate significance characterization. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of two partial system walkdown samples as defined by Inspection Procedure 71111.04-05.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Quarterly Fire Inspection Tours

a. Inspection Scope

The inspectors conducted fire protection walkdowns that were focused on availability, accessibility, and the condition of firefighting equipment in the following risk-significant plant areas:

- Containment Drywell (1A110)

The inspectors reviewed areas to assess if licensee personnel had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant; effectively maintained fire detection and suppression capability; maintained passive fire protection features in good material condition; and had implemented adequate compensatory measures for out of service, degraded or inoperable fire protection equipment, systems, or features, in accordance with the licensees fire plan. The inspectors selected fire areas based on their overall contribution to internal fire risk as documented in the plants Individual Plant Examination of External Events with later additional insights, their potential to affect equipment that could initiate or mitigate a plant transient, or their impact on the plants ability to respond to a security event. Using the documents listed in the attachment, the inspectors verified that fire hoses and extinguishers were in their designated locations and available for immediate use; that fire detectors and sprinklers were unobstructed, that transient material loading was within the analyzed limits; and fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors also verified that minor issues identified during the inspection were entered into the licensees corrective action program. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one quarterly fire-protection inspection sample as defined by Inspection Procedure 71111.05-05.

b. Findings

No findings of significance were identified.

.2 Annual Fire Protection Drill Observation (71111.05A)

a. Inspection Scope

On November 11, 2008, the inspectors observed fire brigade activation for smoke reported in Area 9 of the Auxiliary Building. The observation evaluated the readiness of the plant fire brigade to fight fires. The inspectors verified that the licensee staff identified deficiencies; openly discussed them in a self-critical manner at the drill debrief, and took appropriate corrective actions. Specific attributes evaluated were: (1) proper wearing of turnout gear and self-contained breathing apparatus; (2) proper use and layout of fire hoses; (3) employment of appropriate fire fighting techniques; (4) sufficient firefighting equipment brought to the scene; (5) effectiveness of fire brigade leader communications, command, and control; (6) search for victims and propagation of the fire into other plant areas; (7) smoke removal operations; (8) utilization of pre planned strategies; (9) adherence to the pre planned drill scenario; and (10) drill objectives.

These activities constitute completion of one annual fire-protection inspection sample as defined by Inspection Procedure 71111.05-05.

b. Findings

Introduction. The inspectors identified a Green finding for fire brigade performance deficiencies that were not identified by the licensee during a fire drill critique.

Description. On November 11, 2008, the inspectors observed an unannounced fire drill. The fire drill was started with a report of smoke in the corridor of the 119-foot elevation of the auxiliary building, with the smoke originating from the Division I switchgear room from a fire in a load control center circuit breaker. The inspectors observed operator actions in the control room, along with fire brigade assembly, dress-out, response to the simulated fire in the auxiliary building, and simulated smoke removal activities. The inspectors then observed the drill critique.

Plant personnel identified the following issues:

- The fire brigade was slow to implement overhaul and did not consider re-flash
- Offsite assistance was not requested by the fire brigade or control room

The inspectors identified the following additional performance issues:

- The fire brigade leader had to be prompted to perform a review of fire propagation
- The fire brigade leader did not properly brief the fire brigade prior to entry into the auxiliary building
- The fire brigade leader did not communicate fire pre-plan specifics such as area hazards and plan of attack to the fire brigade team, resulting in team members going to the wrong fire hose
- The fire hose was not fully pulled from the hose reel and had kinks in the hose prior to simulating use
- The fire brigade team failed to simulate manually actuating the CO2 fire suppression system
- Simulated smoke removal was not in accordance with the fire pre-plan
- There was no drill controller in the control room resulting in a lack of assessment of the overall control room operators response

The inspectors determined that some aspects of the drill performance and critique were not rigorous. The inspectors provided feedback to plant personnel on the identified performance issues and the inadequate drill evaluation. Plant training personnel documented the performance deficiencies identified by the inspectors in a condition report and issued corrective actions to implement an improvement plan for fire drill performance and fire drill critiques.

Analysis. The inadequate assessment of fire brigade performance during the unannounced fire drill was a performance deficiency. Grand Gulf Nuclear Station Procedure 10-S-03-7, “Fire Protection Training”, Step 6.9.2.e requires that all drills shall be critiqued to determine the effectiveness in meeting drill objectives. The critique that was performed failed to determine the lack of effectiveness of the fire drill. This finding was more than minor because it was associated with the protection against external factors attribute of the reactor safety mitigating systems cornerstone objective attribute, and adversely affected the cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. Findings associated with the onsite manual fire brigade are excluded from Manual Chapter 0609, Appendix F, “Fire Protection Significance Determination Process.” Therefore, in accordance with Manual Chapter 0609, the safety significance was determined by regional management review. Regional management concluded that the finding was of very low safety significance because it reflected fire brigade performance during a training drill, rather than during an actual fire. The cause of this finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program because the licensee failed to have a low enough threshold in identifying performance issues associated with a plant fire drill [P.1(a)].

Enforcement. No violation of regulatory requirements occurred. This finding was entered into the licensees corrective action program as CR-GGN-2008-06522 and is identified as FIN 05000416/2008005-01, Inadequate Fire Drill Critique.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors performed the annual review of licensee programs, verified performance against industry standards, and reviewed critical operating parameters and maintenance records for the residual heat removal heat exchangers. The inspectors verified that: performance tests were satisfactorily conducted for heat exchangers/heat sinks and reviewed for problems or errors; the licensee utilized the periodic maintenance method outlined in EPRI Report NP 7552, "Heat Exchanger Performance Monitoring Guidelines"; the licensee properly utilized biofouling controls; the licensees heat exchanger inspections adequately assessed the state of cleanliness of their tubes; and the heat exchanger was correctly categorized under 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one heat sink inspection sample as defined by Inspection Procedure 71111.07-05.

b. Findings

No findings of significance were identified.

1R08 In-service Inspection Activities (71111.08)

.1 Inspection Activities Other Than Steam Generator Tube Inspection, Pressurized Water Reactor Vessel Upper Head Penetration Inspections, Boric Acid Corrosion Control (71111.08-02.01)

a. Inspection Scope

The inspection procedure requires review of two or three types of nondestructive examination activities and, if performed, one to three welds on the reactor coolant system pressure boundary. Also, review one or two examinations with recordable indications that have been accepted by the licensee for continued service.

The inspectors directly observed the following nondestructive examinations:

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
LPCS	1E21G002W3	UT
Core Spray	B13-N05B-KB	UT

Core Spray	B13-N05B-KC	UT
RHR	1E12G012H19, B-K Weld	MT

During the review and observation of each examination, the inspectors verified that activities were performed in accordance with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requirements and applicable procedures. Indications were compared with previous examinations and dispositioned in accordance with ASME Code and approved procedures. The qualifications of all nondestructive examination technicians performing the inspections were verified to be current.

None of the above observed or reviewed nondestructive examinations identified any recordable indications, and cognizant licensee personnel stated that no recordable indications were accepted by the licensee for continued service.

The inspectors verified, by review, that the welding procedure specifications and the welders had been properly qualified in accordance with ASME Code, Section IX, requirements. The inspectors also verified, through observation and record review, that essential variables for the welding process were identified, recorded in the procedure qualification record, and formed the bases for qualification of the welding procedure specifications. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one sample as defined by Inspection Procedure 71111.08-05.

b. Findings

No findings of significance were identified.

.2 Identification and Resolution of Problems (71111.08-02.05)

a. Inspection scope

The inspection procedure requires review of a sample of problems associated with in-service inspections documented by the licensee in the corrective action program for appropriateness of the corrective actions.

The inspectors reviewed six condition reports, which dealt with in-service inspection activities and found the corrective actions were appropriate. The specific condition reports reviewed are listed in the documents reviewed section. From this review, the inspectors concluded that the licensee has an appropriate threshold for entering issues into the corrective action program and has procedures that direct a root cause evaluation when necessary. The licensee also has an effective program for applying industry-operating experience. Specific documents reviewed during this inspection are listed in the attachment.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

On December 1, 2008, the inspectors observed a crew of licensed operators in the plants simulator to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

- Licensed operator performance
- Clarity and formality of communications
- Ability to take timely actions in the conservative direction
- Prioritization, interpretation, and verification of annunciator alarms
- Correct use and implementation of abnormal and emergency procedures
- Control board manipulations
- Oversight and direction from supervisors
- Ability to identify and implement appropriate technical specification actions and emergency plan actions and notifications

The inspectors compared the operators performance in these areas to pre-established operator action expectations and successful critical task completion requirements. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one quarterly licensed-operator requalification program sample as defined in Inspection Procedure 71111.11.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors evaluated degraded performance issues involving the following risk significant systems:

- Reactor Core Isolation Cooling (E51)

- engineering safety features Electrical Switchgear and Battery Room Ventilation (T46)
- 125V DC Power Supply and Distribution (L21)

The inspectors reviewed events such as where ineffective equipment maintenance has resulted in valid or invalid automatic actuations of engineered safeguards systems and independently verified the licensees actions to address system performance or condition problems in terms of the following:

- Implementing appropriate work practices
- Identifying and addressing common cause failures
- Scoping of systems in accordance with 10 CFR 50.65(b)
- Characterizing system reliability issues for performance
- Charging unavailability for performance
- Trending key parameters for condition monitoring
- Ensuring proper classification in accordance with 10 CFR 50.65(a)(1) or (a)(2)
- Verifying appropriate performance criteria for structures, systems, and components classified as having an adequate demonstration of performance through preventive maintenance, as described in 10 CFR 50.65(a)(2), or as requiring the establishment of appropriate and adequate goals and corrective actions for systems classified as not having adequate performance, as described in 10 CFR 50.65(a)(1)

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified maintenance effectiveness issues were entered into the corrective action program with the appropriate significance characterization. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of three quarterly maintenance effectiveness samples as defined in Inspection Procedure 71111.12-05.

b. Findings

Introduction. The inspectors identified a Green noncited violation of 10 CFR 50.65(a)(2) involving the failure to adequately monitor the performance of the engineering safety features electrical switchgear and battery room ventilation system.

Description. On September 18, 2008, during a baseline inspection review of condition reports and maintenance work orders, the inspectors identified a condition report from March 2004 that had not been evaluated in the maintenance rule database as a maintenance preventable functional failure of a component in the engineering safety features electrical switchgear and battery room ventilation system. The inspectors also determined that the condition had not been screened to be evaluated as an impact to a maintenance rule system. The condition report identified a room cooler that had tripped due to excessive current on the fan motor because an incorrectly sized sheave was installed during previous maintenance. This improper maintenance impacted the maintenance rule program system function for the engineering safety features room cooler fan to start during emergency operations on standby service water. The failure also impacted the maintenance rule program system function to provide engineering safety features room cooler fan control.

The engineering safety features electrical switchgear and battery room ventilation system is currently classified as a maintenance rule (a)(1) system due to multiple failures of Riley temperature switches. However, had the improper fan maintenance been properly evaluated, the system would have been placed in (a)(1) status in 2004, resulting in increased monitoring and goal setting.

Analysis. The inspectors determined that the finding is a performance deficiency because the licensee failed to apply goals and increase the monitoring of a system impacted by improper maintenance. Upon review of Inspection Manual Chapter 0612, Appendix E, Example 7.b, the inspectors determined that this finding was more than minor since the engineering safety features electrical switchgear and battery room ventilation system was not placed in (a)(1) monitoring status in a timely manner. In addition, the finding was more than minor since violations of 10 CFR 50.65(a)(2) necessarily involve degraded system performance, which, if left uncorrected, could become a more significant safety concern. This finding has very low safety significance because the maintenance rule aspect of the finding did not lead to an actual loss of safety function of the system or cause a component to be inoperable, nor did it screen as potentially risk significant due to a seismic, flooding, or severe weather-initiating event.

Enforcement. Title 10 CFR 50.65(a)(1) requires, in part, that licensees shall monitor the performance or condition of system, structures and components within the scope of the rule against licensee-established goals in a manner sufficient to provide reasonable assurance the system, structures and components are capable of fulfilling their intended safety functions. 10 CFR 50.65(a)(2) requires, in part, that the monitoring specified in paragraph (a)(1) is not required where it has been demonstrated the performance or condition of a system, structures and components is being effectively controlled through the performance of appropriate preventive maintenance such that the system, structures and components remains capable of performing its intended function. Contrary to the above, the licensee failed to demonstrate that the performance or condition of the engineering safety features electrical switchgear and battery room ventilation system had been effectively controlled through the performance of appropriate scheduled maintenance. Specifically, the licensee failed to perform proper maintenance of a system component which demonstrated that the performance of the systems were not being effectively controlled and goal setting and monitoring was required. However, because this inspection finding was characterized by the Significance Determination Process as having very low risk significance (Green) and has been entered in the licensees corrective action program as CR-GGN-2008-02219, this violation is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000416/2008005-02, Failure to Monitor Performance of the engineering safety features Electrical Switchgear and Battery Room Ventilation System.

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

a. Inspection Scope

The inspectors reviewed licensee personnel's evaluation and management of plant risk for the maintenance and emergent work activities affecting risk-significant and safety-related equipment listed below to verify that the appropriate risk assessments were performed prior to removing equipment for work:

- Switchyard work performed the week of September 29, 2008 during Refueling Outage 16

- Plant service water pump trip on October 30, 2008

- Auxiliary building ventilation system maintenance on December 8, 2008

The inspectors selected these activities based on potential risk significance relative to the reactor safety cornerstones. As applicable for each activity, the inspectors verified that licensee personnel performed risk assessments as required by 10 CFR 50.65(a)(4) and that the assessments were accurate and complete. When licensee personnel performed emergent work, the inspectors verified that the licensee personnel promptly assessed and managed plant risk. The inspectors reviewed the scope of maintenance work, discussed the results of the assessment with the licensees probabilistic risk analyst or shift technical advisor, and verified plant conditions were consistent with the risk assessment. The inspectors also reviewed the technical specification requirements and inspected portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of three maintenance risk assessments and emergent work control inspection samples as defined by Inspection Procedure 71111.13 05.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following issues:

- Secondary containment requirements during refueling outage

- CR-GGN-2008-05434, standby service water operability due to structural corrosion in the upper cooling tower areas

- CR-GGN-2008-01861, debris in the condensate storage tank potentially affecting the safety related pumps

- CR-GGN-2008-06727, Riley temperature switch failure renders reactor core isolation cooling and residual heat removal A inoperable

The inspectors selected these potential operability issues based on the risk-significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure that technical specification operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the technical specifications and Updated Final Safety Analysis Report to the licensees evaluations, to determine whether the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations. Additionally, the inspectors also reviewed a sampling of corrective action documents to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of four operability evaluations inspection samples as defined in Inspection Procedure 71111.15-05

b. Findings

Introduction. The inspectors identified two examples of a Green noncited violation of 10 CFR Part 50 Appendix B, Criterion V involving a failure to follow procedures, which resulted in inadequate operability evaluations.

Description. The inspectors identified two examples of inadequate operability evaluations. The first example involved an inadequate evaluation of foreign material in the condensate storage tank. The second example involved an inadequate evaluation of the structural integrity of the standby service water cooling towers.

Example #1: Inadequate Evaluation of Foreign Material in the Condensate Storage Tank

The condensate storage tank is the normal water supply for the reactor core isolation cooling system and the high-pressure core spray system, and provides an inventory of water to the reactor vessel in addition to the suppression pool. During Refueling Outage 15, several pieces of metal were discovered on floor of the tank using an underwater camera while cleaning resin from the floor areas. Plant personnel determined that the debris consisted of a small bolt, two pieces of rupture disc, welding slag and a small clip. Out of this material, only one piece of rupture disc was retrieved. While the remaining items could be physically moved by the remote controlled vacuum device, they could not be removed from the tank using the available equipment. The materials were estimated to be approximately ten feet from the high-pressure core spray and reactor core isolation cooling pump suction flow path. An operability evaluation was performed to document that the debris would not be within the zone of influence for the pump suction flow and thus would not impact the function of the pumps.

On April 16, 2008, a subsequent condition report was written to document that a seismic event could move the debris, which had not been evaluated in the previous operability. Plant personnel performed an operability evaluation and after consultation with seismic engineers, concluded that a seismic event could move the debris into the zone of influence of the pump suction piping. The operability evaluation was based on an assumption that the performance of the high-pressure core spray and reactor core isolation cooling pumps would not be impacted since “the debris would pass through the pump impellers.”

The inspectors reviewed the operability evaluation and concluded that there was no technical basis to show that the foreign material would not cause damage to the pumps. Because of the inspectors questions, plant personnel queried pump vendors on the effects of the identified debris on the function of the pumps. The vendors stated that the material could potentially affect pump operability and function. The inspectors concluded that the assumption in the operability evaluation failed to have a technical basis as required by the plant operability evaluation procedure.

Following this discovery, plant personnel wrote another condition report to document that the operability evaluation did not correctly characterize the potential for moving debris on the bottom of the condensate storage tank during a seismic event. As a result, an engineering evaluation was performed that characterized the water as having no velocity at the bottom of the tank. The inspectors challenged this evaluation and plant personnel reevaluated the seismically induced water velocity to better characterize the movement of water and the induced flow around the debris during a seismic event. The inspectors reviewed the evaluation and noted that the engineering evaluation considered the piece of metal rupture disk to be flat, a non-conservative assumption that would reduce the drag of the suction flow. This was assumed by plant personnel even though previous condition report documentation identified the pieces as “bent” or “mangled.” The engineering evaluation was revised again to include possible lift forces from the water velocity. The inspectors reviewed the revised evaluation and noted that the coefficient of friction between the metal and the floor of the condensate storage tank was a non-conservative value. Plant personnel were ultimately able to provide reasonable assurance of operability by showing that other conservatisms in the calculation bounded the non-conservative coefficient of friction.

Example #2: Inadequate Evaluation of the Structural Integrity of the Standby Service Water Cooling Towers.

On October 3, 2008, during Refueling Outage 16, the inspectors discovered severe corrosion of components in the safety-related cooling towers. Specifically, the distribution beam structural support posts used to support the fill material had severe corrosion and pitting on the visible surface of the posts. The posts extend approximately seven feet below the cooling tower fill, and the condition of the portion of the posts hidden beneath the fill was unknown. The visible, upper portion of the support posts had significant pitting, and subsequent ultrasonic testing ultimately determined that the corrosion had reduced the wall thickness below the ASME minimum wall thickness requirements.

To prove structural operability during a seismic event, plant personnel performed an evaluation of the cooling tower structural design. The inspectors reviewed the evaluation and noted that the calculation only considered loss of structural supports from a single beam. The inspectors concluded that this calculation provided an inadequate technical basis as required by the plant operability evaluation procedure. Plant personnel documented the error in a condition report and reevaluated the operability. Using conventional calculation methods, plant personnel were not able to show that the structure would meet the seismic design criteria. The plant requested an outside organization develop a finite element model to perform a more precise loading analysis. The model included additional rebar that had not been included in the original design calculation, but had been detailed in the original cooling tower structural design drawings. Using the additional rebar in conjunction with the finite element model, plant personnel were able to demonstrate that the structure would remain operable during a design basis seismic event.

Analysis. The failure to implement station procedures was a performance deficiency. This finding was more than minor because the failure to perform adequate operability evaluations, if left uncorrected, could become a more significant safety concern. Using the Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, this finding was of very low safety significance since it did not result in a loss of operability, nor did it screen as potentially risk significant due to a seismic, flooding, or severe weather-initiating event. The cause of this finding has a crosscutting aspect in the area of human performance associated with decision making because licensee personnel failed to use conservative assumptions and did not verify the validity of the underlying assumptions used in making safety-significant decisions [H.1(b)].

Enforcement. Criterion V, "Instructions, Procedures and Drawings," of Appendix B to 10 CFR Part 50 states, in part, that activities affecting quality shall be prescribed by documented instructions and shall be accomplished in accordance with those instructions. On both April 17, 2008 and October 9, 2008, plant engineers failed to accomplish activities affecting quality in accordance with documented instructions. Specifically, Section 5.4[2] of EN-OP-104, "Operability Determinations," Revision 3, required operability evaluations to provide a technical basis for each item in the detailed problem statements per Step 5 of Attachment 9.5 of the procedure. Contrary to the above, on April 17, 2008, plant engineers failed to provide a technical basis for the assumption that the foreign material in the condensate storage tank would not damage the high-pressure core spray and reactor core isolation cooling pumps. On October 9, 2008, plant engineers failed to provide a technical basis for degraded support posts in the standby service water cooling towers. Because these violations were of very low safety significance and were entered in the corrective action program as CR-GGN-2008-05685 and CR-GGN-2008-06044, these violations are being treated as noncited violations consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000416/2008005-03, Two Examples of Inadequate Operability Evaluations.

1R18 Plant Modifications (71111.18)

a. Inspection Scope

The inspectors reviewed the following permanent plant modification to verify that the safety functions of important safety systems were not degraded:

- Implementation of slow start capability of the Division 1 and 2 diesel generators

The inspectors reviewed key affected parameters associated with energy needs, materials/replacement components, timing, heat removal, control signals, equipment protection from hazards, operations, flow paths, pressure boundary, ventilation boundary, structural, process medium properties, licensing basis, and failure modes for the modification listed above. The inspectors verified that modification preparation, staging, and implementation did not impair emergency/abnormal operating procedure actions, key safety functions, or operator response to loss of key safety functions; postmodification testing will maintain the plant in a safe configuration during testing by verifying that unintended system interactions will not occur, systems, structures and components performance characteristics still meet the design basis, the appropriateness of modification design assumptions, and the modification test acceptance criteria will be met; and licensee personnel identified and implemented appropriate corrective actions associated with permanent plant modifications. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one sample for permanent plant modifications as defined in Inspection Procedure 71111.18-05

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed the following postmaintenance activities to verify that procedures and test activities were adequate to ensure system operability and functional capability:

- Division 2 Diesel Generator Loss of Site Power test following a permanent modification change
- Intermediate Range Monitor E functional test following troubleshooting for erratic response
- Division 1 Diesel Generator Loss of Site Power/Loss of Coolant Accident tests following a permanent modification change
- Digital feed water postmaintenance test following permanent modification change to the power supply
- Scram time testing following the replacement of 15 control rod drive mechanisms
- Standby Service Water Train A Fan B postmaintenance test after replacement of the pillow block bearing
- E12-F009 Residual Heat Removal Common Suction valve for shutdown cooling postmaintenance test following the rebuilding of the valve

The inspectors selected these activities based upon the structure, system, or components ability to affect risk. The inspectors evaluated these activities for the following (as applicable):

- The effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed

- Acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate

The inspectors evaluated the activities against the technical specifications, the Updated Final Safety Analysis Report, 10 CFR Part 50 requirements, licensee procedures, and NRC generic communications to ensure that the test results adequately ensured that the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed corrective action documents associated with postmaintenance tests to determine whether the licensee was identifying problems and entering them in the corrective action program and that the problems were being corrected commensurate with their importance to safety. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of seven postmaintenance testing inspection samples as defined in Inspection Procedure 71111.19 05.

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope

The inspectors reviewed the outage safety plan and contingency plans for the refueling outage, conducted September 21 through October 23, 2008, to confirm that licensee personnel had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. During the refueling outage, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities, which are listed below:

- Configuration management, including maintenance of defense-in-depth, is commensurate with the outage safety plan for key safety functions and compliance with the applicable technical specifications when taking equipment out of service

- Clearance activities, including confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing

- Installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication, accounting for instrument error

- Status and configuration of electrical systems to ensure that technical specifications and outage safety-plan requirements were met, and controls over switchyard activities

- Monitoring of decay heat removal processes, systems, and components

- Verification that outage work was not impacting the ability of the operators to operate the spent fuel pool cooling system

- Reactor water inventory controls, including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss

- Controls over activities that could affect reactivity

- Maintenance of secondary containment as required by the technical specifications

- Refueling activities, including fuel handling and sipping to detect fuel assembly leakage

- Startup and ascension to full power operation, tracking of startup prerequisites, walkdown of the drywell (primary containment) to verify that debris had not been left which could block emergency core cooling system suction strainers, and reactor physics testing

- Licensee identification and resolution of problems related to refueling outage activities

Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one refueling outage and other outage inspection sample as defined in Inspection Procedure 71111.20-05.

b. Findings

.1 Failure to Correct Leaking Reactor Water Cleanup System Primary Containment Isolation Valves

Introduction. The inspectors identified a Green noncited violation of 10 CFR Part 50 Appendix B, Criterion XVI, involving the failure to correct leaking reactor water cleanup system primary containment isolation valves.

Description. On September 29, 2008, during Refuelling Outage 16, plant personnel were performing local leak rate testing of the reactor water cleanup system backwash containment penetration. During testing, they determined that the backwash receiving tank transfer isolation valves had exceeded their allowable leakage rate by greater than ten times the allowable limits. The cause of the failure was attributed to resin material collecting in the seats of valves.

The inspectors performed a review of condition reports and determined these valves had failed tests in previous outages. The following listed condition reports document the sites attempts to correct the issue:

- CR-GGN-2005-04189/04975, documented that primary containment isolation valves had failed local leak rate testing during Refuelling Outages 13 and 14. Prior to the failures in Refuelling Outage 14, the site had developed a new method to flush the seats of the valves during backwashing, but had not implemented the procedure change until late in Cycle 14. The site believed that the late implementation of the procedure change was a contributing cause of the valves test failure. They concluded that the new change would work in significantly reducing or eliminating the potential for system material collecting on the seats of the valves.

- CR-GGN-2007-01472, documented that primary containment isolation valves had failed local leak rate testing during Refuelling Outage 15. The site documented that they had flushed the valve seats, which resulted in passing the subsequent leak test. There were no other documented corrective actions.

From this review of condition reports, the inspectors concluded that the methodology for flushing these valves was a long-standing problem that had been previously identified by the licensee. Further review led the inspectors to determine that corrective actions taken by the site to correct the excessive leakage through the valves were ineffective. The site has implemented a new process to more effectively flush the valves. The new procedure change involves flushing the isolation valves by closing and then opening them with flow through the system.

<u>Analysis</u>. The performance deficiency involved the failure to correct excessive leakage from the reactor water cleanup system primary containment isolation valves. The finding was more than minor because it was associated with systems, structures, and components and the barrier performance attribute of the reactor safety barrier integrity cornerstone, and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers would protect the public from the radionuclide releases caused by accident or events. Using Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the finding was determined to be of very low safety significance (Green) since it did not represent an actual open pathway in the physical integrity of the containment system. The cause of this finding has a crosscutting aspect in the area of human performance associated with resources in that the licensee failed to take actions to minimize long-standing equipment issues associated with excessive leakage from primary containment isolation valves [H.2(a)].

<u>Enforcement.</u> 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," states, in part, that conditions adverse to quality are promptly corrected. Contrary to the above, plant personnel did not implement appropriate corrective actions in timely manner following the discovery of excessive leakage from containment isolation valves. Because the finding was of very low safety significance and has been entered into the licensees corrective action program as Condition Report CR-GGN-2008-05139, this violation is being treated as a noncited violation, consistent with section VI.A of the NRC Enforcement Policy. NCV 05000416/2008005-04, Failure to Correct Leaking Reactor Water Cleanup System Primary Containment Isolation Valves.

.2 Trip of a Reactor Recirculation Pump During Pump Up-shift to Fast Speed Due to Ineffective Corrective Actions

Introduction. The inspectors reviewed a self-revealing Green finding involving a recirculation pump trip during pump up-shift to fast speed due to ineffective corrective actions.

Description. On October 25, 2008, while up shifting the reactor recirculation Pump A to fast speed, the pump tripped resulting in a decrease in reactor power. The plant investigated the trip and found the third phase of Breaker 252-1103C (CB4A) indicated an instantaneous over-current trip condition. The plant had recently replaced the recirculation Pump A motor during Refuelling Outage 16. The engineering department conducted an investigation and determined that the instantaneous over-current trip for the breaker was set at 43.2 amps and should have been set at 44 to 46 amps, and attributed the difference to setpoint drift. The protective relay was calibrated and the recirculation Pump A was restarted in fast speed without any issues.

The inspectors performed a review of condition reports and determined that the reactor recirculation Pump B had tripped following motor replacement for the same reason in September 2007. The inspectors reviewed the apparent cause evaluation and determined that the extent of condition was inadequate. The sites review had stated that although these devices were used throughout the plant in numerous applications, there had not been any recently tripped devices due to setpoint drift. The licensee checked the other two phases of the recirculation Pump B, found these devices were within calibration, and concluded the trip was an isolated incident. The licensee did not check the trip devices on the recirculation Pump A breaker, nor did they initiate action to check the calibration of the devices following the scheduled replacement of the Pump A motor in the next refuelling outage.

The site issued condition report CR-GGN-2008-06269 to determine why their process did not direct them to perform calibration checks during the replacement of the recirculation Pump A motor during Refuel Outage 16.

Analysis. The performance deficiency involved the failure of plant personnel to properly implement the corrective action program. Specifically, plant personnel were directed to perform an extent of condition per section 4.0 Step 4.c of EN-LI-119, “Apparent Cause Evaluation Process,” to ensure similar occurrences would be prevented. Contrary to this, plant personnel failed to perform an adequate extent of condition, which resulted in the trip of recirculation Pump A during up-shift to fast speed. 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 Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the finding was determined to be of very low safety significance (Green) since it did not contribute to loss of function of mitigating equipment. The cause of this finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program in that the licensee failed to perform a thorough evaluation of a problem that resulted in a plant transient such that the resolution properly addressed the cause and extent of condition [P.1(c)].

Enforcement. No violation of regulatory requirements occurred. This finding was entered into the licensees corrective action program as CR-GGN-2008-06269 and is identified as FIN 05000416/2008005-05, “Trip of a Reactor Recirculation Pump During Pump Up-shift to Fast Speed Due to Ineffective Corrective Actions.”

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the Updated Final Safety Analysis Report, procedure requirements, and technical specifications to ensure that the four surveillance activities listed below demonstrated that the systems, structures, and/or components tested were capable of performing their intended safety functions. The inspectors either witnessed or reviewed test data to verify that the significant surveillance test attributes were adequate to address the following:

- Preconditioning
- Evaluation of testing impact on the plant
- Acceptance criteria
- Test equipment
- Procedures
- Jumper/lifted lead controls
- Test data
- Testing frequency and method demonstrated technical specification operability
- Test equipment removal
- Restoration of plant systems
- Fulfillment of ASME Code requirements
- Updating of performance indicator data
- Engineering evaluations, root causes, and bases for returning tested systems, structures, and components not meeting the test acceptance criteria were correct
- Reference setting data
- Annunciators and alarms setpoints.

The inspectors also verified that licensee personnel identified and implemented any needed corrective actions associated with the surveillance testing.

- September 25, 2008, Containment isolation valve local leak-rate test, drywell cooling Valve P71-F150, Procedure 06-ME-1M61-V-0002

- September 28, 2008, Containment isolation valve local leak-rate test, shutdown cooling suction Valve E12-F008, Procedure 06-ME-1M61-V-0002

- October 17, 2008, Reactor vessel in-service leak test, Procedure 03-1-01-6

- October 21, 2008, reactor core isolation cooling pump low pressure flow verification test, Procedure 06-OP-1E51-C-0005, in-service test

- October 27, 2008, Containment integrated leak rate test, Procedure 06-ME-1M10-O-0002

Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of five surveillance testing inspection samples as defined in Inspection Procedure 71111.22-05.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

a. Inspection Scope

The inspector performed an in-office review of Revision 60 to the Grand Gulf Nuclear Station Emergency Plan, implemented July 17, 2008. This revision replaced use of the State of Mississippi Department of Health mobile radiological laboratory with sample analysis performed at the Department of Health's radiological laboratory in Jackson, Mississippi, eliminated the Technical Engineering Manager and Emergency Operations Facility engineering emergency response organization positions and reassigned their duties to the Technical Manager and to Technical Support Center engineers, reassigned the Accident Assessment Engineer from the Emergency Operations Facility to the Technical Support Center, updated Letters of Agreements with offsite agencies, updated the emergency planning zone Evacuation Time Estimate to September 2007, reformatted the Fission Product Barrier Matrix in the licensees emergency action levels, and made other minor administrative corrections.

The revision was compared to its previous revision, to the criteria of NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1, and to the standards in 10 CFR 50.47(b) to determine if the revision adequately implemented the requirements of 10 CFR 50.54(q). This review was not documented in a Safety Evaluation Report and did not constitute an approval of the licensees changes; therefore, these revisions are subject to future inspection.

These activities constitute completion of one sample as defined in Inspection Plan 71114.04-05.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational and Public Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope

This area was inspected to assess the licensees' 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 licensees 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
- Controls (surveys, posting, and barricades) of three radiation, high radiation, or airborne radioactivity areas
- Radiation work permits procedures, engineering controls, and air sampler locations
- Conformity of electronic personal dosimeter alarm set points with survey indications and plant policy; workers knowledge of required actions when their electronic personnel dosimeter noticeably malfunctions or alarms
- Barrier integrity and performance of engineering controls in two airborne radioactivity areas
- Adequacy of the licensees internal dose assessment for any actual internal exposure greater than 50 millirem committed effective dose equivalent
- Physical and programmatic controls for highly activated or contaminated materials (non-fuel) stored within spent fuel and other storage pools
- 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
- Licensee actions in cases of repetitive deficiencies or significant individual deficiencies

- Radiation work permit briefings and worker instructions
- Adequacy of radiological controls, such as required surveys, radiation protection job coverage, and contamination control during job performance
- Dosimetry placement in high radiation work areas with significant dose rate gradients
- Changes in licensee procedural controls of high dose rate - high radiation areas and very high radiation areas
- Controls for special areas that have the potential to become very high radiation areas during certain plant operations
- 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 21 of the required 21 samples as defined in Inspection Procedure 71121.01-05.

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 reasonably achievable (ALARA). The inspector used the requirements in 10 CFR Part 20 and the licensees' 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 as defined in Inspection Procedure 71121.02-05.

b. <u>Findings</u>

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Mitigating Systems Performance Index - Emergency AC Power System

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index - Emergency AC Power System performance indicator for the period from the fourth quarter 2007 through the third quarter 2008. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, was used. The inspectors reviewed the licensees operator narrative logs, mitigating systems performance index derivation reports, issue reports, event reports and NRC integrated inspection reports for the period of October 2007 through September 2008 to validate the accuracy of the submittals. The inspectors reviewed the mitigating systems performance index component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensees issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator, and none was identified. Specific documents reviewed are described in the attachment to this report.

These activities constitute completion of the mitigating systems performance index emergency ac power system sample as defined by Inspection Procedure 71151-05.

b. Findings

No findings of significance were identified.

.2 Mitigating Systems Performance Index - High Pressure Injection Systems

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index - High Pressure Injection Systems performance indicator for the period from the fourth quarter 2007 through the third quarter 2008. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, was used. The inspectors reviewed the licensees operator narrative logs, issue reports, mitigating systems performance index derivation reports, event reports and NRC integrated inspection reports for the period of October 2007 through September 2008 to validate the accuracy of the submittals. The inspectors reviewed the mitigating systems performance index component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensees issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator, and none was identified. Specific documents reviewed are described in the attachment to this report.

These activities constitute completion of the mitigating systems performance index high-pressure injection system sample as defined by Inspection Procedure 71151-05.

b. Findings

No findings of significance were identified

.3 Mitigating Systems Performance Index - Heat Removal System

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index - Heat Removal System performance indicator for the period from the fourth quarter 2007 through the third quarter 2008. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, was used. The inspectors reviewed the licensees operator narrative logs, issue reports, event reports, mitigating systems performance index derivation reports, and NRC integrated inspection reports for the period of October 2007 through September 2008 to validate the accuracy of the submittals. The inspectors reviewed the mitigating systems performance index component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensees issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator, and none was identified. Specific documents reviewed are described in the attachment to this report.

These activities constitute completion of the mitigating systems performance index heat removal system sample as defined by Inspection Procedure 71151-05.

b. Findings

No findings of significance were identified.

.4 Mitigating Systems Performance Index - Residual Heat Removal System

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index - Residual Heat Removal System performance indicator for the period from the fourth quarter 2007 through the third quarter 2008. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, was used. The inspectors reviewed the licensees operator narrative logs, issue reports, mitigating systems performance index derivation reports, event reports and NRC integrated inspection reports for the period of October 2007 through September 2008 to validate the accuracy of the submittals. The inspectors reviewed the mitigating systems performance index component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensees issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator, and none was identified. Specific documents reviewed are described in the attachment to this report.

These activities constitute completion of the mitigating systems performance index residual heat removal system sample as defined by Inspection Procedure 71151-05.

b. Findings

No findings of significance were identified.

.5 Mitigating Systems Performance Index - Cooling Water Systems

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index - Cooling Water Systems performance indicator for the period from the fourth quarter 2007 through the third quarter 2008. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, was used. The inspectors reviewed the licensees operator narrative logs, issue reports, mitigating systems performance index derivation reports, event reports and NRC integrated inspection reports for the period of October 2007 through September 2008 to validate the accuracy of the submittals. The inspectors reviewed the mitigating systems performance index component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensees issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator, and none was identified. Specific documents reviewed are described in the attachment to this report.

These activities constitute completion of the mitigating systems performance index cooling water system sample as defined by Inspection Procedure 71151-05.

b. Findings

No findings of significance were identified.

.6 Occupational Radiological Occurrences

a. Inspection Scope

The inspectors sampled licensee submittals for the Occupational Radiological Occurrences performance indicator for the period from the first quarter 2008 through third quarter 2008. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in NEI Document 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 5, was used. The inspectors reviewed the licensees assessment of the performance indicator for occupational radiation safety to determine if indicator related data was adequately assessed and reported. To assess the adequacy of the licensees performance indicator data collection and analyses, the inspectors discussed with radiation protection staff, the scope and breadth of its data review, and the results of those reviews. The inspectors independently reviewed electronic dosimetry dose rate and accumulated dose alarm and dose reports and the dose assignments for any intakes that occurred during the time period reviewed to determine if there were potentially unrecognized occurrences.

These activities constitute completion of the occupational radiological occurrences sample as defined by Inspection Procedure 71151-05.

b. Findings

No findings of significance were identified.

.7 Radiological Effluent Technical Specifications/Offsite Dose Calculation Manual Radiological Effluent Occurrences

a. Inspection Scope

The inspectors sampled licensee submittals for the Radiological Effluent Technical Specifications/Offsite Dose Calculation Manual Radiological Effluent Occurrences performance indicator for the period from the first quarter 2008 through third quarter 2008. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in NEI Document 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 5, was used. The inspectors reviewed the licensees issue report database since this indicator was last reviewed to identify any potential occurrences such as unmonitored, uncontrolled, or improperly calculated effluent releases that may have impacted offsite dose. Additionally, the inspectors reviewed the licensees historical 10 CFR 50.75(g) file and selectively reviewed the licensees analysis for discharge pathways resulting from a spill, leak, or unexpected liquid discharge focusing on those incidents which occurred over the last few years.

These activities constitute completion of the radiological effluent technical specifications/offsite dose calculation manual radiological effluent occurrences sample as defined by Inspection Procedure 71151-05.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness, Public Radiation Safety, Occupational Radiation Safety, and Physical Protection

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

As part of the various baseline inspection procedures discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that they were being entered into the licensees corrective action program at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. The inspectors reviewed attributes that included the complete and accurate identification of the problem; the timely correction, commensurate with the safety significance; the evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent of condition reviews, and previous occurrences reviews; and the classification, prioritization, focus, and timeliness of corrective actions. Minor issues entered into the licensees corrective action program because of the inspectors observations are included in the attached list of documents reviewed.

These routine reviews for the identification and resolution of problems did not constitute any additional inspection samples. Instead, by procedure, they were considered an integral part of the inspections performed during the quarter and documented in Section 1 of this report.

b. Findings and Observations

No findings of significance were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

In order to assist with the identification of repetitive equipment failures and specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensees’ corrective action program. The inspectors accomplished this through review of the stations daily corrective action documents.

The inspectors performed these daily reviews as part of their daily plant status monitoring activities and, as such, did not constitute any separate inspection samples.

b. Findings and Observations

No findings of significance were identified.

.3 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a review of the licensees corrective action program and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors focused their review on repetitive equipment issues, but also considered the results of daily corrective action item screening discussed in Section 4OA2.2 above, licensee trending efforts, and licensee human performance results. The inspectors nominally considered the 6-month period of May 1, 2008, through December 1, 2008, although some examples expanded beyond those dates where the scope of the trend warranted. The inspectors reviewed the following issues:

- Maintenance risk evaluation
- Maintenance rule program implementation
- Missed surveillances
- Inadequate electrical connections during maintenance
- Condition report downgrading and classification
- Fire protection equipment reliability

The inspectors also included issues documented outside the normal corrective action program in major equipment problem lists, repetitive and/or rework maintenance lists, departmental problem/challenges lists, system health reports, quality assurance audit/surveillance reports, self-assessment reports, and Maintenance Rule assessments. The inspectors compared and contrasted their results with the results contained in the licensees corrective action program trending reports. Corrective actions associated with a sample of the issues identified in the licensees trending reports were reviewed for adequacy.

These activities constitute completion of one semi-annual trend inspection sample as defined in Inspection Procedure 71152-05.

b. Findings and Observations

The inspectors evaluated the licensees corrective action program trending methodology and observed that the licensee had performed detailed reviews of developing issues. In the past six months, several condition reports were written to evaluate emerging trends. In addition to those trends identified by the licensee, the inspectors noted the following:

Maintenance Risk Evaluations: The inspectors reviewed several issues identified by both plant personnel and NRC inspectors where maintenance risk was not evaluated during the work planning process. In addition, several issues were identified where operators failed to correctly identify emerging conditions as impacts to station risk. These adverse trends had not been identified in a plant condition report or station quality assurance trend reports.

Maintenance Rule Evaluations: During review of plant equipment failures, the inspectors noted a recent increasing trend in maintenance rule functional failure evaluations that were not performed, poorly performed, or performed with inappropriate results. The inspectors noted that many of these substandard evaluations were a result of poorly written and ambiguous explanations of system functions. Plant personnel have documented problems with maintenance rule functional failure evaluations in several condition reports; however, no adverse trend had been identified in a plant condition report or station quality assurance trend reports.

Inadequate Electrical Connections: During a review of condition reports and maintenance work orders, the inspectors noted an adverse trend of equipment problems identified during postmaintenance testing. Several of these problems were due to wiring errors not detected by schematic checks or other self-checking techniques. Plant personnel had identified the adverse trend of wiring deficiencies in Condition Report CR-GGNS-2008-05910.

.4 Annual Sample: Review of Operator Workarounds

a. Scope

The inspectors evaluated the licensees implementation of their process used to identify, document, track, and resolve operational challenges. Inspection activities included, but were not limited to, a review of the cumulative effects of the operator workarounds on system availability and the potential for improper operation of the system, for potential impacts on multiple systems, and on the ability of operators to respond to plant transients or accidents.

The inspectors performed a review of the cumulative effects of operator workarounds. The documents listed in the attachment were reviewed to accomplish the objectives of the inspection procedure. The inspectors reviewed both current and historical operational challenge records to determine whether the licensee was identifying operator challenges at an appropriate threshold, had entered them into their corrective action program and proposed or implemented appropriate and timely corrective actions, which addressed each issue. Reviews were conducted to determine if any operator challenge could increase the possibility of an Initiating Event, if the challenge was contrary to training, required a change from long-standing operational practices, or created the potential for inappropriate compensatory actions. Additionally, all temporary modifications were reviewed to identify any potential effect on the functionality of Mitigating Systems, impaired access to equipment, or required equipment uses for which the equipment was not designed. Daily plant and equipment status logs, degraded instrument logs, and operator aids or tools being used to compensate for material deficiencies were also assessed to identify any potential sources of unidentified operator workarounds.

These activities constitute completion of one operator workarounds annual inspection sample as defined in Inspection Procedure 71152-05.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up (71153)

.1 Severe Corrosion of Components in the Safety-Related Standby Service Water Cooling Towers

a. Inspection Scope

During Refueling Outage 16, the inspectors discovered corrosion of components in the safety-related Division I standby service water-cooling towers on October 3, 2008. The components included the distribution beam structural support posts used to support the fill material, wall hangers that support the ends of the spray piping headers, the piping restraint system used to secure the spray piping laterals to the distribution beams, and the cooling tower spray piping. Plant personnel evaluated the condition of the standby service water spray piping restraint system, which resulted in the replacement of all the U-bolt restraints in the Division I cooling towers. In addition, wall hanger supports were cleaned and recoated and the degraded piping was cleaned, ultrasonically tested for pit depth, and recoated. The extent of condition was evaluated and a similar condition was found in the Division II standby service water cooling towers. Plant personnel implemented the same corrective actions to restore the condition of the Division II cooling towers. Documents reviewed in this inspection are listed in the Attachment.

b. Findings

.1 Failure to Prevent Recurrence of Standby Service Water Corrosion

Introduction. The inspectors identified a Green noncited violation of 10 CFR Part 50 Appendix B, Criterion XVI, involving a failure to take corrective actions to prevent recurrence of corrosion in piping hangers, piping supports, and piping in the standby service water basin cooling towers.

Description. On October 3, 2008, during Refueling Outage 16, the inspectors discovered corrosion of components in the safety-related Division I standby service water-cooling towers. The components included the distribution beam structural support posts used to support the fill material, wall hangers that support the ends of the spray piping headers, the piping restraint system used to secure the spray piping laterals to the distribution beams, and the cooling tower spray piping. The inspectors discovered U-bolts on the piping restraint system completely severed, wall hanger support nuts with greater than 50 percent material loss, and corrosion and pitting underneath blistered coating on the standby service water spray piping system. Plant personnel evaluated the condition of the spray piping restraint system, which resulted in the replacement of all the U-bolt restraints in the Division I standby service water-cooling towers. In addition, wall hanger supports were cleaned and recoated and the degraded piping was cleaned, ultrasonically tested for pit depth, and recoated. The extent of condition was evaluated and a similar condition was found in the Division II cooling towers. Plant personnel implemented the same corrective actions to restore the condition of the Division II cooling towers.

The inspectors determined that significant corrosion on the same components in the standby service water-cooling tower cells had been previously identified by plant personnel during a ten-year in-service inspection on October 3, 1993. At the time, plant personnel determined this to be a significant degraded condition of a safety related system, requiring replacement of the piping and associated supports.

To determine past operability, plant personnel evaluated varying conditions of the piping restraint system, including the bounding case, which considered all restraints non-functional. A linear elastic analysis model was used to evaluate the spray piping system stresses for faulted loads. The results showed that the bounding piping stresses would not exceed the pressure retaining boundary operability limits of the spray piping.

Analysis. The inspectors determined that the failure to take corrective actions to prevent recurrence of standby service water piping support corrosion was a performance deficiency. This finding was more than minor because the corrosion represented a degrading condition that if left uncorrected could become more significant safety concern. The finding was also more than minor because it was associated with the protection against external factors attribute of the reactor safety mitigating systems cornerstone, and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using the Manual Chapter of 0609, "Significance Determination Process," Phase 1 Worksheet, this finding was of very low safety significance since it did not represent an actual loss of safety function, nor did it screen as potentially risk significant due to a seismic, flooding, or severe weather-initiating event.

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," states, in part, that in the case of a significant condition adverse to quality, measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition. Contrary to the above, plant personnel did not implement corrective actions to preclude repetition of a significant condition adverse to quality in the form of severe corrosion of structures inside the standby service water cooling towers. Specifically, on October 3, 2008, the inspectors identified the piping support degradation, which was a repeat occurrence with the same root cause as degradation identified by plant personnel on October 3, 1993. Because the finding was of very low safety significance and has been entered into the corrective action program as Condition Report CR-GGN-2008-05434, this violation is being treated as a noncited violation, consistent with section VI.A of the NRC Enforcement Policy. NCV 05000416/2008005-06, Failure to Prevent Recurrence of Standby Service Water Corrosion.

.2 Untimely Corrective Actions Following Identification of Degrading Standby Service Water Supports

Introduction. The inspectors identified a Green noncited violation of 10 CFR Part 50 Appendix B, Criterion XVI, involving the failure to take timely corrective actions for corrosion on distribution beam structural support posts in the standby service water basin cooling towers.

Description. On October 3, 2008, during Refueling Outage 16, the inspectors discovered severe corrosion of components in the Division I safety-related standby service water-cooling towers. The components included the distribution beam structural support posts used to support the fill material, wall hangers that support the ends of the spray piping headers, the piping restraint system used to secure the spray piping laterals to the distribution beams, and the cooling tower spray piping. The inspectors discovered

U-bolts on the piping restraint system completely severed, wall hanger support nuts with greater than 50 material loss, and corrosion and pitting underneath blistered coating on the standby service water spray piping system. One of the subsets of the components, the distribution beam structural support posts used to support the fill material, were found with severe corrosion and pitting on the visible surface of the posts. The distribution beam structural support posts are carbon steel pipes six inches in diameter, approximately twelve feet in length, connected from the distribution piping support beam at the top of the cooling tower to the concrete lower fill support beams at the bottom of the cooling tower. The posts extend approximately seven feet below the cooling tower fill. The visible, upper portion of the support posts had significant pitting, and ultrasonic testing ultimately determined that the corrosion had reduced the wall thickness below the ASME minimum wall thickness requirements.

The inspectors determined that the corrosion on the support posts had originally been identified by plant personnel on August 9, 2005. The inspectors reviewed documents evaluating these corrosion issues with the support post but found no documentation identifying any other corrosion issues with other components inside the standby service water cooling towers. The inspectors additionally reviewed a site plan to inspect these support posts on quarterly bases during the cycle. No documentation of the degraded condition of these other components was documented by the site during the quarterly inspections.

An operability evaluation performed in 2005 required operability to be reevaluated after engineering developed “an understanding of corrosion rate.” The inspectors reviewed the operability evaluations and noted that the next operability evaluation was performed over two years later on December 23, 2007. The revised operability evaluation used a rate of corrosion provided by engineering that was evaluated to be “conservative and would assure operability of the support posts until the upcoming refueling outage.” The revised operability evaluation properly designated the corrosion as a significantly degraded, non-conforming condition; however, the inspectors concluded this classification should have been applied upon discovery of the condition in 2005.

The inspectors reviewed the engineering documentation that determined the corrosion rate and specified the inspection methodology and criteria for rework and replacement of the support posts. The instructions stated that the support posts were to be sandblasted, inspected via ultrasonic testing to determine thickness, and then either replaced or coated with paint. When the inspectors entered into the cooling towers, the support posts in one cooling tower cell had already been sandblasted and painted. The inspectors discovered that the painting was done prior to performing the required ultrasonic testing, contrary to the requirements detailed in the engineering documentation. The inspectors relayed this information immediately to plant management, and plant personnel were directed to perform ultrasonic testing of the uncoated support posts in the second cooling tower cell. However, during the interim, the support posts in the second cell were coated, preventing an evaluation of uncoated support posts. Plant personnel developed a methodology to determine the minimum wall thickness of the coated support posts, and determined that the corrosion on the posts had exceeded the ASME minimum wall thickness requirements. Plant personnel were able to demonstrate using a finite element analysis that even with a complete failure of the support posts; the structure would remain functional during a safe shutdown earthquake.

In addition, the corroded wall hanger supports identified by the inspectors were cleaned and recoated and the degraded piping was cleaned, ultrasonically tested for pit depth, and recoated. The extent of condition was evaluated and a similar condition was found in the Division II cooling towers. Plant personnel implemented the same corrective actions to restore the condition of the Division II cooling towers. Plant personnel evaluated the spray piping system using a linear elastic analysis model. The results showed that the bounding piping stresses would not exceed the pressure retaining boundary operability limits of the spray piping.

Analysis. The inspectors determined that the failure to promptly identify and correct the degraded standby service water cooling tower structures was a performance deficiency. The finding was more than minor because it was associated with the protection against external factors attribute of the reactor safety mitigating systems cornerstone, and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the finding was determined to be of very low safety significance since it did not represent an actual loss of a safety function of the standby service water cooling towers, nor did it screen as potentially risk significant due to a seismic, flooding, or severe weather-initiating event. The cause of this finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program because licensee personnel failed to identify issues completely, accurately, and in a timely manner commensurate with their safety significance. [P.1(a)].

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," states, in part, that conditions adverse to quality are promptly identified and corrected. Contrary to the above, plant personnel did not promptly identify and correct a condition adverse to quality. Specifically, plant personnel failed to identify severely corroded components in the standby service water towers and did not implement timely corrective actions following the discovery of degraded cooling tower support posts. Because the finding was of very low safety significance and has been entered into the corrective action program as Condition Report CR-GGN-2008-05434, this violation is being treated as a noncited violation, consistent with section VI.A of the NRC Enforcement Policy. NCV 05000416/2008005-07, Untimely Corrective Actions Following Identification of Degrading standby service water Supports.

.2 Reactor Scram Due to a Loss of Reactor Feedwater Turbine

Inspection Scope

The inspectors responded to a reactor scram due to total loss of feedwater during plant start-up from a refuel outage with the plant at approximately 15 percent power on October 23, 2008. Operators restored reactor water level using reactor core isolation cooling, and then transitioned to the condensate system via the start-up level control valve to maintain reactor water level. The appropriate emergency operating, off-normal event and integrating operating procedures were entered to mitigate the transient with all systems responding as designed. Site personnel investigating the scram determined that an operator had incorrectly performed actions for the reactor feed Pump B turbine on the Pump A control switches at a local panel. This resulted in a total loss of steam pressure to the running reactor feed pump causing a total loss of feedwater flow and low reactor water level scram. Documents reviewed in this inspection are listed in the Attachment.

a. Findings

.1 Automatic Reactor Scram Caused by an Operator Inadvertently Closing the Steam Supply Valves to Reactor Feed Pump Turbine

Introduction. The inspectors reviewed a self-revealing Green finding involving an automatic reactor scram caused by an operator inadvertently closing the steam supply valves to a reactor feed pump turbine.

Description. On October 23, 2008, a reactor scram occurred due to total loss of feedwater during plant start-up from a refuel outage with the plant at approximately 15 percent power. The inspectors responded to the control room to observe scram recovery actions. Operators restored reactor water level using reactor core isolation cooling, and then transitioned to the condensate system via the start-up level control valve to maintain reactor water level.

Site personnel investigating the scram determined that an operator had been sent out to perform restoration actions on reactor feed Pump B following over-speed testing. The operator arrived at the local panel, which contained controls for both Train A, and Train B reactor feed pump turbines and incorrectly performed the actions for Pump B on the Pump A control switches. In the process of manipulating switches, he inadvertently closed the steam supply valves to the reactor feed Pump A turbine. This resulted in a total loss of steam pressure to the only running reactor feed pump, which resulted in a total loss of feedwater flow.

Operations management issued standing orders requiring a peer check prior to manipulating controls at the feed pump panel and other plant panels with similar error traps. The operations department has since placed protective covers over critical switches that could cause plant transients at these panels.

Analysis. The performance deficiency involved the failure of an operator to ensure he was performing valve manipulations on the correct feed pump. Specifically, the operator was directed per Attachment 1 step 7.13 of 04-1-03-N21-7, “Reactor Feed Pump ‘B’ Over-speed Trip Test,” to shutdown the reactor feed Pump B turbine. Contrary to this, the operator manipulated switches on reactor feed Pump A, resulting in an isolation of the steam supply to the Pump A turbine. The finding was more than minor because it was associated with the initiating events cornerstone attribute of human 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 Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the inspectors concluded that an evaluation was required by the regional senior reactor analyst, because the finding impacted both the initiating event and mitigating systems cornerstones. The senior reactor analyst performed a Phase 3 analysis assuming the feed pumps were not recoverable and determined that that conditional core damage probability was 8.7E-7, with no impact to large early release frequency. As a result, the issue was of very low safety significance (Green). The cause of this finding has a crosscutting aspect in the area of human performance associated with work practices because the operator failed to use proper self-checking techniques while performing actions to place feed Pump B in the standby lineup [H.4(a)].

Enforcement. No violation of regulatory requirements occurred. This finding was entered into the licensees corrective action program as CR-GGN-2008-06195 and is identified as FIN 05000416/2008005-08, Automatic Reactor Scram Caused by an Operator Inadvertently Closing the Steam Supply Valves to Reactor Feed Pump Turbine.

.2 Failure to Properly Monitor Plant Parameters to Control Reactor Coolant System Cooldown Rate

Introduction. The inspectors identified a Green finding involving the failure to demonstrate proper monitoring of plant parameters to control reactor coolant system cooldown rate to within expected management standards.

Description. On October 23, 2008, the plant experienced a reactor scram from approximately 15 percent power during plant start-up from a refuelling outage due to a total loss of feedwater. The inspectors responded to the control room to observe scram recovery actions.

The control room supervisor had entered the emergency operating procedures after the scram occurred. He directed a reactor operator to use the reactor core isolation cooling system to restore reactor water level to the normal band. Reactor water level had decreased due to total loss of feedwater injection. The reactor operator used reactor core isolation cooling with suction from the condensate storage tank and raised water level approximately 70 inches. Reactor pressure decreased at a faster rate than operators expected due to low decay heat levels and the injection of relatively cold water to the reactor vessel from the condensate storage tank. The control room supervisor transitioned from the emergency operating procedures to the scram recovery integrated operating instruction and ordered the control room operators to remove the steam jet air ejector from service to reduce the rate of reactor pressure decrease. The control room supervisor did not give a pressure band after pressure decreased below the low end of the emergency operating procedure band of 800 psig or assigned a licensed reactor operator to monitor reactor pressure during the event. Additionally, operators did not report a decreasing reactor pressure trend to the control room supervisor.

The inspectors observed the operators response to the scram and recognized that the plant was approaching the procedure limit of cooldown rate of 90 degrees per hour. The inspectors brought the decreasing reactor pressure to the shift managers attention and questioned whether the operators were considering shutting the main steam isolation valves to prevent exceeding the cooldown limit. The shift manager questioned the control room supervisor about the actions the operators were taking to limit the cooldown rate. The shift manager then told the control room supervisor the operators would have to close the main steam isolation valves to limit the cooldown rate. The control room supervisor gave the order to close the inboard main steam isolation valves. Reactor pressure had decreased to approximately 362 psig.

During the post scram review, plant personnel recognized that they had not exceeded reactor coolant cooldown rate for the one-hour period. The site did determine they had cooled down 103.2 degrees in 49 minutes, based on steam dome temperatures but had only cooled down 84 degrees in one hour based on reactor recirculation loop temperatures. Additionally, plant personnel performed a review of all the data and determined that they had not exceeded cooldown rates for the vessel metal temperatures; therefore, the reactor coolant system was acceptable for continued operation.

The inspectors conducted a follow-up review of the event and determined based on discussions with plant management, licensed operators and licensed operator training personnel, that the operator crew had not demonstrated proper control of cooldown rate following the reactor scram. The operating crew allowed reactor pressure to decrease to below 450 psig without taking actions to close the main steam isolation valves. Operations management expectation is for operators to monitor critical plant parameters such as reactor pressure and take timely actions to limit the potential impact to the plant.

<u>Analysis</u>. The inspectors determined that operators failing to demonstrate proper control of critical parameters are a performance deficiency. Specifically, plant operators are directed by section 4.0 [7] c, m and [11] f, of EN-OP-115, "Conduct of Operations," to monitor critical parameters during abnormal or emergency conditions and to take action to mitigate adverse trends. Contrary to this, the operating crew did not demonstrate proper monitoring or control of reactor pressure. This finding is more than minor because the failure to control critical parameters during transients, if left uncorrected, could become a more significant safety concern. Using the Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the inspectors determined that the finding has very low safety significance (Green) since it did not represent an actual violation of cooldown rate. The cause of this finding has a crosscutting aspect in the area of human performance associated with work practices, because control room supervision failed to maintain proper oversight to ensure reactor coolant cooldown rate was maintained within procedural limits. [H.4(c)].

<u>Enforcement</u>. No violation of regulatory requirements occurred. This finding was entered into the licensees corrective action program as CR-GGN-2008-06201 and is identified as FIN 05000416/2008005-09, "Failure to Properly Monitor Plant Parameters to Control Reactor Coolant System Cooldown Rate."

.3 <u>Turbine Trip and Reactor Scram Due to Failure of the Thyristor Voltage Regulator</u>

a. <u>Inspection Scope</u>

The inspectors responded to a reactor scram due to a turbine/generator trip and turbine control valve fast closure on October 26, 2008. The inspectors verified that the appropriate emergency operating, off-normal event and integrating operating procedures were entered to mitigate the transient with all systems responding as designed. Plant personnel completed troubleshooting of the voltage regulator and determined that two separate problems caused the trip. First, the voltage regulator tripped from automatic mode to manual mode. Second, the manual setpoint was set at a lower value and had not tracked with the automatic setpoint as designed. With the voltage regulator regulating at a lower voltage following the swap, the excitation voltage lowered and caused generator reactivity to drop which resulted in a generator trip signal. Plant personnel determined that the incorrect setpoint tracking was due to a failed raise/lower motor in the manual setpoint card. This card was replaced and tested to ensure proper function. Plant personnel did not determine the cause for the swap to manual, so corrective actions were issued to replace the logic cards in the voltage regulator cabinet and to provide operators with detailed instructions on operation of the voltage regulator in manual mode. Documents reviewed in this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

.4 Plant Service Water Pump Trip

a. Inspection Scope

On October 30, 2008, the inspectors responded to a trip of plant service water radial well Pump J due to a failed level switch. Plant personnel entered the loss of plant service water off-normal event procedure to mitigate the loss of the pump. Plant personnel started that plant service water Pump C and plant service water header pressure remained normal throughout the event. The inspectors determined that plant personnel had taken the appropriate actions and the plant was in a stable condition. Documents reviewed in this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

.5 Reactor Recirculation Pump Trip

a. Inspection Scope

The inspectors responded to a trip of a reactor recirculation Pump B on October 30, 2008. Plant personnel had been raising power and had reached the power level where it was necessary to transfer the reactor recirculation pumps from slow to fast speed. As Pump A transitioned to fast speed, the mechanical seal for the recirculation pump failed to stage properly. Using the system operating instructions, plant personnel returned the pump to slow speed. At this time, recirculation Pump B tripped offline. Plant personnel entered the reduction in recirculation flow off-normal event procedure to mitigate the loss of the pump. The inspectors determined that plant personnel had taken the appropriate actions and the plant was in a stable condition. Documents reviewed in this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

.6 Feedwater Pump Fire

a. Inspection Scope

On November 17, 2008, a building operator identified smoke coming from the feedwater pump room and notified the control room. The resident inspector was in the control room at the time of notification. The control room initiated a fire alarm and dispatched the plant fire brigade to the scene. Plant personnel declared a Notice of an Unusual

Event. The fire originated underneath the turbine near and below the outer turbine bearing. Oil leaking from the lube oil reservoir had drenched the insulation and then ignited due to the high metal temperatures of the turbine housing. The fire brigade initially put out the flames, but the fire re-flashed as more oil leaked out of the reservoir. During the event, the control room reduced power, tripped the reactor feed pump, and shutdown the lube oil supply pumps. Documents reviewed in this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

4OA5 Other Activities

.1 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

During the inspection period, the inspectors performed observations of security force personnel and activities to ensure that the activities were consistent with Grand Gulf Nuclear Station security procedures and regulatory requirements relating to nuclear plant security. These observations took place during both normal and off-normal plant working hours.

These quarterly resident inspector observations of security force personnel and activities did not constitute any additional inspection samples. Rather, they were considered an integral part of the inspectors normal plant status review and inspection activities.

b. Findings

No findings of significance were identified.

.2 (Closed) Implementation of Temporary Instruction 2515/176, "Emergency Diesel Generator Technical Specification Surveillance Requirements Regarding Endurance and Margin Testing"

a. Inspection Scope

The objective of Temporary Instruction 2515/176 was to gather information to assess the adequacy of nuclear power plant emergency diesel generator endurance and margin testing as prescribed in plant-specific technical specifications. The inspectors reviewed the licensees Technical Specifications, procedures, and calculations and interviewed licensee personnel to complete the temporary instruction. The information gathered while completing this temporary instruction was forwarded to the Office of Nuclear Reactor Regulation for further review and evaluation on December 12, 2008.

b. Findings

No findings of significance were identified.

4OA6 Meetings

Exit Meeting Summary

On January 13, 2009, the inspectors presented the inspection results to Mr. J. Browning, General Manager, Plant Operations, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. None was identified.

On October 3, 2008, the inspector presented the Occupational Radiation Safety inspection results to Mr. R. Douet, Vice President, Operations, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

On October 3, 2008, the inspectors presented the In-service inspection results to Mr. R. Douet, Vice President, Operations, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

On November 6, 2008, the inspector conducted a telephonic exit meeting to present the results of the in-office inspection of changes to the licensees' emergency plan to Ms. M. Wilson, Manager, Emergency Preparedness, who acknowledged the findings.

4OA7 Licensee-Identified Violations

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

- Technical Specification 3.4.4 requires main steam relief Valve 1B21F041E to be operable to perform the safety relief function to prevent reactor pressure vessel over-pressurization. Contrary to this requirement, on September 25, 2007, Valve 1B21F041E exceeded Surveillance Requirement 3.4.4.1, which is performed per the in-service testing program. The licensee staff interpreted the ASME code interval to be from installation of the valve in the plant instead of from when the valve was last tested. This resulted in the valve being outside its test interval. This issue was of very low safety significance since it did not result in an actual loss of operability. This issue was documented in the licensees corrective action program per CR-GGN-2007-04715.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel
D. Barfield, Director, Engineering
J. Browning, General Manager, Plant Operations
M. Causey, Maintenance Rule Engineer
K. Christian, Engineering Program Supervisor
A. Cockrum, Site Welding Engineer
R. Collins, Manager, Corrective Actions and Assessments
D. Coulter, Licensing Specialist, Plant Licensing
P. Different, Senior Lead Engineer, Reactor Engineering
R. Douet, Vice President, Operations
B. Edwards, Minority Owner Specialist
R. Gardner, Manger, Maintenance
E. Harris, Manager, Quality Assurance
K. Higginbotham, Manager, Operations
R. Jackson, Licensing Specialist, Plant Licensing
D. Jones, Manager, System Engineering
M. Krupa, Director, Nuclear Safety and Assurance
G. Lantz, Supervisor, Design Engineering
M. Larson, Licensing Engineer
M. McAdory, Senior Operations Instructor
R. McGaha, ISI Engineering
S. Osborn, Licensing
J. Owens, Licensing Specialist, Plant Licensing
C. Perino, Licensing Manager
M. Rohrer, Manager, Component Engineering
J. Smyrl, ISI Engineering, Level III
T. Tankersley, Manager, Training
T. Thornton, Manager, Design Engineering
W. Trichell, Supervisor, Radiation Protection
D. Wilson, Supervisor, Design Engineering
F. Wilson, Manager, Planning, Scheduling and Outages
M. Wilson, Manager, Emergency Preparedness
R. Wilson, Radiation Protection Manager
P. Worthington, Supervisor, Engineering
E. Wright, ALARA Specialist, Radiation Protection

NRC Personnel
W. Walker, Senior Project Engineer
M. Runyan, Senior Reactor Analyst
G. Guerra, Emergency Preparedness Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Opened and Closed

05000416/2008005-01	FIN	Inadequate Fire Drill Critique
05000416/2008005-02	NCV	Failure to Monitor Performance of the engineering safety features Electrical Switchgear and Battery Room Ventilation System
05000416/2008005-03	NCV	Two Examples of Inadequate Operability Evaluations
05000416/2008005-04	NCV	Failure to Correct Leaking Reactor Water Cleanup System Primary Containment Isolation Valves
05000416/2008005-05	FIN	Trip of a Reactor Recirculation Pump During Pump Up-shift to Fast Speed Due to Ineffective Corrective Actions
05000416/2008005-06	NCV	Failure to Prevent Recurrence of Standby Service Water Corrosion
05000416/2008005-07	NCV	Untimely Corrective Actions Following Identification of Degrading standby service water Supports
05000416/2008005-08	FIN	Automatic Reactor Scram Caused by an Operator Inadvertently Closing the Steam Supply Valves to Reactor Feed Pump Turbine
05000416/2008005-09	FIN	Failure to Property Monitor Plant Parameters to Control Reactor Coolant System Cooldown Rate

Closed

TI 2515/176	TI	Emergency Diesel Generator Technical Specification Surveillance Requirements Regarding Endurance and Margin Testing

Discussed

None

LIST OF DOCUMENTS REVIEWED

Section 1RO4: Equipment Alignment

DOCUMENTS

NUMBER	TITLE	REVISION / DATE
04-1-01-P44-1	Plant Service Water/Radial Wells	089
04-1-01-P75-1	Standby Diesel Generator System	076

Section 1RO5: Fire Protection

CONDITION REPORTS

CR-GGN-2008-06522 CR-GGN-2008-06529 CR-GGN-2008-05925
CR-GGN-2008-05578

DOCUMENTS

NUMBER	TITLE	REVISION / DATE
10-S-03-7	Fire Drill Scenario and Score Card	012
10-S-03-2	Fire Protection Procedure Response to Fires	019
ES-01	Electrical Standard For The Installation Of Electrical Raceway	002

OTHER

Fire Pre-Plan A-26, Revision 1
Fire Pre-Plan A-16, Revision 0
Fire Pre-Plan A-07, Containment Building Room 1A110, Revision 1
Fire Pre-Plan C-14, Revision 1
Fire Pre-Plan C-15, Revision 1
Fire Pre-Plan C-15, Revision 3
Drawing E-7093; Raceway Plan Containment Building Area 11 Unit 1; Revision 2
Drawing E-7094; Raceway Plan Containment Building Area 11 Unit 1; Revision 3
Drawing E-7095; Raceway Plan Containment Building Area 11 Unit 1; Revision 2
Drawing E-7096; Raceway Plan Containment Building Area 11 Unit 1; Revision 3
Drawing E-KA7095; Raceway Plan Containment Building Area 11 Unit 1; Revision 4

Section 1RO7: Heat Sink Performance

CONDITION REPORTS

CR-GGN-2008-04914 CR-GGN-2008-05026

DOCUMENT

NUMBER	TITLE	REVISION / DATE
STI-0802	RHR “B” HX Online Thermal Performance Test	000

WORK ORDER

WO150179

OTHER

EN-EP-S-039-G, Testing Standard for Safety-Related Heat Exchangers Cooled by Standby Service Water, Revision 0

Vendor Manual 460002111, TEMA Data Sheets

EC-10620

MSDS – NALCO 73551

Calculation MC-Q1111-08011, RF16 Wall Thickness Data Evaluation, Revision 0

QC Inspection Detail Report, Inspection No.: 2008-00003

RHR HX Initial Inspection Guidelines

Section 1R08: In-Service Inspection Activities

CONDITION REPORTS

CR-GGN-2008-04838 CR-GGN-2008-04813 CR-GGN-2007-04217
CR-GGN-2007-02954 CR-GGN-2007-02341 CR-GGN-2008-00951

PROCEDURES

NUMBER	TITLE	REVISION / DATE
CEP-NDE-0404	Manual Ultrasonic Testing of Ferritic Piping Welds	3
CEP-NDE-0731	Magnetic Particle Examination (MT) for ASME XI	2

WELDING ON PRESSURE BOUNDARY

SYSTEM/COMPONENT	DESCRIPTION	WR NUMBER
E12F017B	Fabrication and Installation of Socket Welds	080089
P52F122	Fabrication and Installation of Valve per EC-7716	080166
P11F004	Fabrication and Installation of butt Weld	080095
E22	Fabrication and Installation of Large Pipe Weld	080117

Section 1R11: Licensed Operator Requalification Program

OTHER

GSMS-LOR-EXPEC; Operator Simulator Performance; Attachment III

GSMS-LOR-00223; ECCS Suction Leak/Adjacent Sump Check Valve Failure; Revision 01

Section 1R12: Maintenance Effectiveness

CONDITION REPORTS

CR-GGN-2008-05908	CR-GGN-2008-05658	CR-GGN-2007-04913
CR-GGN-2007-04770	CR-GGN-2007-04767	CR-GGN-2007-04709
CR-GGN-2007-04706	CR-GGN-2007-04703	CR-GGN-2007-04321
CR-GGN-2008-04121	CR-GGN-2007-03776	CR-GGN-2007-03773
CR-GGN-2008-03751	CR-GGN-2007-03072	CR-GGN-2007-02963
CR-GGN-2007-02615	CR-GGN-2008-02614	CR-GGN-2008-02085
CR-GGN-2008-01545	CR-GGN-2008-00567	

OTHER

Rolling 18 Month Unavailability E51

Maintenance Rule Database, reactor core isolation cooling System E51

Maintenance Rule Program a(1) Evaluation, Condensate and Refueling Water Storage and Transfer (P11) System

Section 1R13: Maintenance Risk Assessment and Emergent Work Controls

CONDITION REPORT

CR-GGN-2008-05408

Section 1R15: Operability Evaluations

CONDITION REPORTS

CR-GGN-2008-06044 CR-GGN-2008-01861 CR-GGN-2007-01032

DOCUMENTS

NUMBER	TITLE	REVISION / DATE
03-1-01-5	Refueling	119
01-S-06-50	Control of Fuel Services Operations	003
02-S-01-17	Control of Limiting Conditions for Operation	115
06-OP-1T10-M-0001	Secondary Containment Penetration Isolation Monthly Check	105

OTHER

EC 11163

EC11197

Drawing M-1414, “Yard Piping Sections and Details Unit 1,” Revision 9

Drawing GG-1-FIG-OP-P1100, “Equipment, Piping, and Valves at Condensate and Refueling Water Storage Tanks” Figure 5

Drawing GG-1-FIG-OP-P1100, “Condensate Storage Tank” Figure 6

GG UFSAR 6.3-10a, LDC 05013

GG UFSAR 3.7-8, Revision 2

GIN-2008/00317, “Response to NRC Question Concerning Condensate Storage Tank Failure during a Seismic Event”

Section 1R18: Plant Modifications

CONDITION REPORTS

CR-GGN-2008-05838	CR-GGN-2008-05753	CR-GGN-2008-05722
CR-GGN-2008-05186	CR-GGN-2008-04937	

OTHER

ECT-2048-04 Functional Test-Division 2 Standby Diesel Generator Final Acceptance Testing, Woodward 2301A/EGB-35P/DRU Governor System

ECT-2048-01 Functional Test-Division 2 Standby Diesel Generator Bench Testing and Setup, Woodward 2301A Governor and Digital Reference Unit

ECT-2408-02 Functional Test – Division 2 Standby Diesel Generator Logic Functional Check

ECT-2048-03 Functional Test – Division 2 Standby Diesel Generator On-Engine Setup, Woodward 2301A Governor and Digital Reference Unit

EC-6113

EC-6114

Section 1R19: Postmaintenance Testing

CONDITION REPORTS

CR-GGN-2008-06057	CR-GGN-2008-05952	CR-GGN-2008-05951
CR-GGN-2008-05906	CR-GGN-2008-05392	

DOCUMENTS

NUMBER	TITLE	REVISION / DATE
06-ME-1M61-V-0003	Local Leak Rate Test – Low Pressure Water	104
06-OP-1E12-C-0012	RHR A Shutdown Cooling Mode Valve Test	110
06-OP-1P75-R-0003	SDG 11, 18 Month Functional Test – Test No. 6 – Div 1 LOP/LOCA Test	113
06-OP-1P75-R-0003	SDG 11, 18 Month Functional Test – Test No. 4 – Loss of Offsite Power	113
06-OP-1C51-V-0002	IRM Functional Test	106

DOCUMENTS

NUMBER	TITLE	REVISION / DATE
06-RE-SC11-V-0402	Control Rod Scram Testing – Individual Scram – Manual Analysis Method (Section 5.4)	116
06-OP-1P75-R-0004	Standby Diesel Generator 12: 18 Month Functional Test"	114

WORK ORDERS

WO00131969	WO00169164-01	WO51680811-01
WO51659411-01	WO51513661-01	WO51513659-01
WO51515080-01	WO00131492-01	

OTHER

Batch Scram Time Data Beginning 20081014 and Ending 20081104
Clearance 1C17-1 P41-002-1P41-C003B
ECT-4538-01-000, “Test Change Notice 001”

Section 1R20: Refueling and Other Outage Activities

CONDITION REPORTS

CR-GGN-2008-06539	CR-GGN-2008-06319	CR-GGN-2008-06269
CR-GGN-2008-06238	CR-GGN-2008-06110	CR-GGN-2008-06100
CR-GGN-2008-05904	CR-GGN-2008-05902	CR-GGN-2008-05819
CR-GGN-2008-05797	CR-GGN-2008-05658	CR-GGN-2008-05499
CR-GGN-2008-05465	CR-GGN-2008-05392	CR-GGN-2008-05381
CR-GGN-2008-05294	CR-GGN-2008-05294	CR-GGN-2008-05166
CR-GGN-2008-05139	CR-GGN-2008-05124	CR-GGN-2008-05111
CR-GGN-2008-04983	CR-GGN-2005-04975	CR-GGN-2007-04576
CR-GGN-2005-04189	CR-GGN-2007-01472	CR-GGN-2000-01123
CR-GGN-2002-00930		

NUMBER	TITLE	REVISION / DATE
04-1-01-G33-1	Reactor Water Cleanup	137
EN-LI-119	Apparent Cause Evaluation (ACE) Process	007

OTHER

Safety Assessment of the RF16 Outage Schedule, July 23, 2008, Revision 0

Calculation Sheet MC-OSP64-86058, Revision 61

Standard Number ES-01, Revision 2

Drywell Closeout Sheet, December 11, 2008

NRC Information Notice Number 91-41; Potential Problems with the Use of Freeze Seals, June 27, 1991

NRC Information Notice 2006-26; Failure of Magnesium Rotors in Motor-Operated Valve Actuators, November 20, 2006

SK-MJH-100488, GGNS FTS Cable Drive Reeving Diagram

Grand Gulf Nuclear Station RF16 Refueling Outage Daily Updates, September 24 – October 22, 2008

Grand Gulf QA Observation and O2C Summary Reports RF16, September 21 – October 14, 2008.

Section 1R22: Surveillance Testing

CONDITION REPORT

CR-GGN-2008-06066

DOCUMENTS

NUMBER	TITLE	REVISION / DATE
06-OP-1E51-C-0005	RCIC Pump Low Pressure Flow Verification Test	106
03-1-01-6	Integrated Operating Instruction Reactor Vessel In-Service Leak Test	117
06-ME-1M10-O-0002	Containment Integrated Leak Rate Test	104
06-ME-1M10-O-0003	Drywell Bypass Leakage Rate	103
06-ME-1M61-V-0002	Local Leak Rate Test -AIR (using Graftel Model 9623-7 Leak Rate Monitor)	105

WORK ORDER

WO00163539

Section 2OS1: Access Controls to Radiologically Significant Areas

CONDITION REPORTS

CR-GGN-2008-3770	CR-GGN-2008-4198	CR-GGN-2008-4594
CR-GGN-2008-4803	CR-GGN-2008-4808	CR-GGN-2007-4808
CR-GGN-2008-4812	CR-GGN-2008-4821	CR-GGN-2008-5349

DOCUMENTS

NUMBER	TITLE	REVISION / DATE
01-S-08-1	Administration of the Grand Gulf Nuclear Station Radiation Protection Program	104
01-S-08-2	Exposure and Contamination Control	117
08-S-02-50	Radiological Surveys and Surveillances	115
EN-RP-100	Radworker Expectations	0
EN-RP-101	Access Control for Radiologically Controlled Areas	2
EN-RP-108	Radiation Protection Posting	5
EN-RP-105	Radiation Work Permits	1

RADIATION WORK PERMITS

20081052	20081054	20081400	20081505	20081508	20081512
20081514	20081516	20081530			

Section 2OS2: ALARA Planning and Controls

AUDITS, SELF-ASSESSMENTS, AND SURVEILLANCES

DOCUMENTS

NUMBER	TITLE	REVISION / DATE
01-S-08-2	Exposure and Contamination Control	117
EN-RP-102	Radiological Control	000
EN-RP-110	ALARA Program,	002

Section 2OS2: ALARA Planning and Controls

AUDITS, SELF-ASSESSMENTS, AND SURVEILLANCES

DOCUMENTS

NUMBER	TITLE	REVISION / DATE
EN-RP-150	Radiography and X-Ray Testing	003

MISCELLANEOUS

2008 – 2012 Five Year ALARA Plan

Section 4OA1: Performance Indicator Verification

CONDITION REPORT

CR-GGN-07096

DOCUMENT

NUMBER	TITLE	REVISION / DATE
GGNS-SA-06-002	GGNS MSPI Basis Document and Supporting Information Documentation	004

OTHER

Access Control to Radiologically Significant Areas and PI Verification, GLO-2008-0032, June 26, 2008

NRC Performance Indicator Technique/Data Sheet, Emergency AC Power (EDG), Fourth Quarter 2007 – Third Quarter 2008

NRC Performance Indicator Technique/Data Sheet, High Pressure Injection Fourth Quarter 2007 – Third Quarter 2008

NRC Performance Indicator Technique/Data Sheet, Heat Removal (RCIC), Fourth Quarter 2007 – Third Quarter 2008

NRC Performance Indicator Technique/Data Sheet, Residual Heat Removal Fourth Quarter 2007 – Third Quarter 2008

NRC Performance Indicator Technique/Data Sheet, Cooling Water Support, Fourth Quarter 2007 – Third Quarter 2008

Emergency Diesel/Division I & II Unavailability Hours, October 2007 – September 2008

EPIX P75 Summary, October 2007 – September 2008

HPCS/E22 System Unavailability Hours, October 2007 – September 2008

HPCS SSW/P41 System Division III Unavailability Hours, October 2007 – September 2008

Emergency Diesel/Division III Unavailability Hours, October 2007 – September 2008

EPIX E22 Summary, October 2007 – September 2008

RCIC/E51 System Unavailability Hours, October 2007 – September 2008

EPIX E51 Summary, October 2007 – September 2008

RHR/E12 Division I & II Unavailability Hours, October 2007 – September 2008

EPIX E12 Summary, October 2007 – September 2008

SSW/P41 System Division I & II Unavailability Hours, October 2007 – September 2008

EPIX P41 Summary, October 2007 – September 2008

Section 4OA2: Identification and Resolution of Problems

CONDITION REPORTS

CR-GGN-2008-05910 CR-GGN-2008-05174 CR-GGN-2008-05018
CR-GGN-2008-04990

PROCEDURES

NUMBER	TITLE	REVISION / DATE
EN-OP-104	Operability Evaluation	003
EN-OP-115	Conduct of Operations	6

OTHER

Grand Gulf Nuclear Station Operations Workarounds, November 14, 2008

Section 4OA3: Event Follow-Up

CONDITION REPORTS

CR-GGN-2008-06584	CR-GGN-2008-06576	CR-GGN-2008-06571
CR-GGN-2008-06568	CR-GGN-2008-06318	CR-GGN-2008-06317
CR-GGN-2008-06305	CR-GGN-2008-06246	CR-GGN-2008-06239
CR-GGN-2008-06238	CR-GGN-2008-06201	CR-GGN-2008-06197
CR-GGN-2008-06195		

PROCEDURES

NUMBER	TITLE	REVISION / DATE
01-S-06-26	Post-Trip Analysis	017
01-S-06-5	Event Notification EN# 44595	108
10-S-03-5	Plant Fire Report	101
03-1-01-4	Integrated Operating Instruction Scram Recovery	110
04-1-01-N21-1	Feedwater System	062
04-1-03-N21-7	Reactor Feed Pump ‘B’ Overspeed Trip	025
06-IC-1C34-R-0001	Reactor Vessel Water Level High (Level 9) MT/RFPT Trip Calibration	105
EN-OP-111	Operational Decision-Making Issue (ODMI) Process	3

OTHER

CA-00001 CR-GGN-2008-0654 “Reactor Feedwater Pump Turbine B Fire Risk”
Bases Figure B 3.3.1.1-1, Reactor Vessel Water Level, B3.3-30a
Entergy External Talk Points, “Grand Gulf Declares Unusual Event”
Single Trend Point B21N091A.C88
Alarm Response Instruction, 04-1-02-1H13-P680-3A-D9, Revision 181

Section 4OA5: Other Activities

PROCEDURES

NUMBER	TITLE	REVISION / DATE
06-OP-1P75-R-0003	Standby Diesel Generator 11: 18 Month Functional Test	Revision 113

Section 4OA5: Other Activities

PROCEDURES

NUMBER	TITLE	REVISION / DATE
06-OP-1P75-R-0004	Standby Diesel Generator 11: 18 Month Functional Test	Revision 114
06-OP-1P81-R-0001	HPCS Diesel Generator: 18 Month Functional Test	Revision 115

CALCULATIONS

NUMBER	TITLE	REVISION / DATE
E-DCP 82/5020-1	Transient Diesel Generator Loading (LOP/LOCA)	Revision A

Section 4OA7: Licensee-Identified Violations

CONDITION REPORT

CR-GGN-2007-04715

LIST OF ACRONYMS USED

ALARA	As Low As Reasonably Achievable
ASME	American Society of Mechanical Engineers
GGNS	Grand Gulf Nuclear Station
NCV	noncited violation
NOUE	Notice of Unusual Event
CFR	Code of Federal Regulations