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

January 14, 2003

William A. Eaton, Vice President
Operations - Grand Gulf Nuclear Station
Entergy Operations, Inc.
P.O. Box 756
Port Gibson, Mississippi 39150

SUBJECT: GRAND GULF NUCLEAR STATION - NRC INSPECTION REPORT 50-416/02-05

Dear Mr. Eaton:

On December 28, 2002, 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 8, 2003, with Mr. J. Roberts and other members of your staff.

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

This report documents three findings of very low safety significance (Green) which were determined to involve violations of NRC requirements. However, because of the very low safety significance and because they were entered into your corrective action program, the NRC is treating these three findings as noncited violations (NCVs) consistent with Section VI.A of the NRC Enforcement Policy. If you contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Grand Gulf Nuclear Station facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

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

Sincerely,

/RA/

William D. Johnson, Chief
Project Branch A
Division of Reactor Projects

Docket: 50-416
License: NPF-29

Enclosure: Inspection Report 50-416/02-05
w/Attachment: Supplemental Information

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

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

Winston & Strawn
1400 L Street, N.W. - 12th Floor
Washington, DC 20005-3502

Director
Division of Solid Waste Management
Mississippi Department of Natural
Resources
P.O. Box 10385
Jackson, Mississippi 39209

President, District 1
Claiborne County Board of Supervisors
P.O. Box 339
Port Gibson, Mississippi 39150

General Manager
Grand Gulf Nuclear Station
Entergy Operations, Inc.
P.O. Box 756
Port Gibson, Mississippi 39150

The Honorable Richard Ieyoub
Attorney General
Department of Justice
State of Louisiana
P.O. Box 94005
Baton Rouge, Louisiana 70804-9005

Office of the Governor
State of Mississippi
Jackson, Mississippi 39201

Mike Moore, Attorney General
Frank Spencer, Asst. Attorney General
State of Mississippi
P.O. Box 22947
Jackson, Mississippi 39225

Dr. Brian W. Amy
State Health Officer
State Board of Health
P.O. Box 1700
Jackson, Mississippi 39215

Robert W. Goff, Program Director
Division of Radiological Health
Mississippi Dept. of Health
P.O. Box 1700
Jackson, Mississippi 39215-1700

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

Director, Nuclear Safety
and Regulatory Affairs
Entergy Operations, Inc.
P.O. Box 756
Port Gibson, Mississippi 39150

Electronic distribution by RIV:
Regional Administrator **(EWM)**
DRP Director **(ATH)**
DRS Director **(DDC)**
Senior Resident Inspector **(TLH4)**
Branch Chief, DRP/A **(WDJ)**
Senior Project Engineer, DRP/A **(CJP)**
Staff Chief, DRP/TSS **(PHH)**
RITS Coordinator **(NBH)**
Dale Thatcher **(DFT)**
Scott Morris **(SAM1)**
GG Site Secretary **(MJS)**

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RWDeese	TLHoeg	JMKeeton	TWPruett	WDJohnson
E-WDJohnson	***E-WDJohnson***	***/RA/***	***JBNicholas***	***/RA/***
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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-416
License:	NPF-29
Report No:	50-416/02-05
Licensee:	Entergy Operations, Inc.
Facility:	Grand Gulf Nuclear Station
Location:	Waterloo Road Port Gibson, Mississippi 39150
Dates:	September 29 through December 28, 2002
Inspectors:	T. L. Hoeg, Senior Resident Inspector D. R. Carter, Health Physicist R. W. Deese, Resident Inspector J. S. Dodson, Health Physicist/Regional Operations Officer J. M. Keeton, Project Engineer R. E. Lantz, Senior Emergency Preparedness Inspector M. P. Shannon, Senior Health Physicist
Approved By:	W. D. Johnson, Chief Reactor Projects Branch A Division of Reactor Projects
Attachment:	Supplemental Information

SUMMARY OF FINDINGS

IR 05000416/2002-005; Entergy Operations, Inc., 09/29/02 - 12/28/02; Grand Gulf Nuclear Station; Refueling and Outage Activities; Identification and Resolution of Problems.

The report covered a 13 week period of inspection by resident inspectors and regional reactor safety inspectors. Three Green noncited violations were identified. The significance of any findings are indicated by their color (Green, White, Yellow, or Red) using IMC 0609 "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. <u>Inspector Identified and Self-Revealing Findings</u>

Cornerstone: Initiating Events

- Green. A noncited violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified for failure to establish appropriate instructions for restoration of a reactor recirculation Loop B decontamination flange which resulted in improper torquing of flange bolting and degrading a reactor coolant system (RCS) pressure boundary. This issue was documented in the licensee's correction action program as CR-GGN-2002-1988.

  The noncited violation is greater than minor because it affected the initiating events cornerstone objective of limiting the likelihood of an initiating event in the form of a loss of coolant from the flanged pressure boundary. The finding was of very low safety significance because although the bolts were overtorqued and would have been exposed to RCS pressure, the bolts were replaced by the licensee prior to taking to RCS to operating pressure due to inspector intervention (Section 1R20).

Cornerstone: Mitigating Systems

- Green. A noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Actions," was identified for inadequate corrective actions which resulted in operating the residual heat removal (RHR) system heat exchanger outlet Valve (E12-F003A) beyond its optimum throttling range causing small bore piping failures. This issue was documented in the licensee's correction action program as CR-GGN-2002-1779.

  This self-revealing noncited violation is greater than minor because it affected the mitigating system cornerstone objective of equipment reliability, in that operation of this valve beyond its optimum throttling capability would lead to system small bore piping failures. The finding was of very low safety significance because, all other remaining emergency core cooling systems remained available (Section 4OA2).

- Green. A noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified for inadequate design controls which resulted in a pressure locking design modification being completed without provisions for adequate piping supports resulting in a small bore piping failure. This issue was documented in the licensee's correction action program as CR-GGN-2002-1779.

This self-revealing noncited violation is greater than minor because it affected the Mitigating System Cornerstone objective of equipment reliability, in that the inadequate design of the pressure locking piping modification allowed cyclic fatigue to cause a through wall crack of the piping and ultimately complete failure of the small bore piping. The finding was of very low safety significance because all other remaining emergency core cooling systems remained available (Section 4OA2).

B. Licensee Identified Findings

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

Report Details

Summary of Plant Status: The Grand Gulf Nuclear Station (GGNS) began this inspection period in Mode 6 during scheduled refueling outage (RFO) Number 12. The plant completed RFO 12 on October 4, 2002, and returned to 100 percent reactor power on October 6. On October 19, reactor power was lowered to 50 percent to perform troubleshooting of the Train A reactor feedwater pump turbine control circuit. The plant then returned to 100 percent rated thermal power of 3833 megawatts (MW) until October 26 when the licensee implemented a power uprate to a new rated thermal power of 3898 MW. The plant was then operated at or near 100 percent rated thermal power except for periodic planned power reductions for monthly control rod exercising and periodic control rod pattern adjustments until November 7 when power was reduced to 68 percent to perform planned maintenance on main steam isolation Valve B21F028C control circuitry. Power was returned to 100 percent on November 8 and remained there until November 15 when power was reduced to 50 percent when the Train B main circulating water pump tripped due to an electrical failure. Power was returned to 100 percent on November 16 where it remained until November 30 when power was again reduced about 70 percent to locate and suppress a leaking fuel pin. Power was returned to 100 percent on November 30 and remained there throughout this inspection period.

1. REACTOR SAFETY

Initiating Events, Mitigating Systems, Barrier Integrity [Reactor - R]

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

Prior to the onset of cold weather conditions, the inspectors reviewed GGNS's readiness to operate under freezing conditions. Equipment Performance Instruction 04-1-03-A30-1, "Cold Weather Protection," Revision 13, was reviewed and site walkdowns were performed by the inspectors to verify the licensee had made the required preparations for cold weather. The inspection also included a detailed review of; (1) the standby service water system, (2) the fire protection water system, and (3) the instrument air system to ensure they were protected from freezing temperatures.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Partial System Walkdowns (71111.04)

a. Inspection Scope

The inspectors performed partial system walkdown inspections and reviews of a train in each of three systems important to reactor safety in order to verify the operability of the systems. The inspectors reviewed system operating instructions, system valve and breaker lineups, operator logs, and system control room indications. The inspectors also verified valves, breakers, and control circuits were in their required positions for operability. The following systems were inspected:

- High pressure core spray system
- Instrument air system
- Division III emergency diesel generator

b. Findings

No findings of significance were identified.

.2 Semi-Annual Complete System Walkdown (71111.04S)

a. Inspection Scope

During November 4-5, 2002, the inspectors performed a complete walkdown of the Division I emergency diesel generator system to determine if there were any discrepancies between the actual equipment alignment versus what was procedurally required. During the walkdown, System Operating Instruction 04-1-01-P75-1, "Standby Diesel Generator System," Revision 62, Surveillance Procedure 06-OP-1P75-V-0013, "Standby Diesel Generator (SDG) 11 Operability Verification," Revision 103, and Drawing M-1070B, "Standby Diesel Generator System," Revision 30, were used by the inspectors to verify major diesel generator components were correctly labeled and aligned. The inspectors also reviewed open condition reports on the system for any deficiencies that could affect the ability of the system to perform its design function. Documentation associated with control room deficiencies, temporary modifications, operator workarounds, and items tracked by plant engineering were also reviewed to assess their collective impact on system operation.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors reviewed area fire plans and performed walkdowns of seven plant areas to assess the materiel condition and operational status of fire detection, suppression systems and equipment; the materiel condition of fire barriers; and the control of transient combustibles. Specific risk-significant plant areas included:

- Containment ventilation equipment room, Room 1A405
- Division II emergency diesel generator room, Room 1D303
- High pressure core spray pump room, Room 1A109
- Residual heat removal Train A heat exchanger room, Room 1A102
- Residual heat removal Train B pump room, Room 1A105
- Standby liquid control pump area, Area 1A512

- Upper cable spreading relay room, Room 0C703

b. Findings

No findings of significance were identified.

1R06 Flood Protection (71111.06)

a. Inspection Scope

The inspectors reviewed one sample of the licensee's internal flooding protection features and general flood protection measures for the high pressure core spray system room. The inspectors performed a walkdown of the area reviewing internal flooding vulnerabilities including the following: water tight door operation; room high water level alarm system; sealing of electrical conduits at or near floor level; and potential sources of internal flooding. The inspectors also reviewed the protective features and procedures for mitigating the impact of any flooding.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11Q)

a. Inspection Scope

On October 30, 2002, the inspectors observed two scenarios during one session of licensed operator requalification training activities in the simulator to assess the licensee's effectiveness in conducting the requalification program and to verify that licensed individuals received the appropriate level of training required to maintain their licenses. The first scenario in the observed training was GG-1-SMS-LOR-00178-04, Part 5, "Loss of Vacuum with Anticipated Transient Without Scram Using Level/Power Control." The second scenario observed was GG-1-SMS-LOR-00178-04, Part 2, "Loss of Coolant Accident with Loss of Offsite Power Forcing Emergency Depressurization at the Top of Active Fuel." The inspectors also observed the post-training critiques conducted by the training instructors and the shift manager to verify that weak areas observed during simulator operations were appropriately identified for additional training.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12Q)

a. Inspection Scope

The inspectors reviewed performance-based problems involving three selected in-scope structures, systems, or components (SSCs) to assess the effectiveness of the

Maintenance Rule Program. Reviews focused on: (1) proper Maintenance Rule scoping in accordance with 10 CFR 50.65; (2) characterization of failed SSCs; (3) safety significance classifications; (4) 10 CFR 50.65 (a)(1) and (a)(2) classifications; and, (5) the appropriateness of performance criteria for SSCs classified as (a)(2), and goals and corrective actions for SSCs classified as (a)(1). The inspectors reviewed the most recent system health reports and system functional failures for the last two years. The following conditions were reviewed:

- Division I residual heat removal system
- Division II emergency diesel generator
- 125 Volt DC battery chargers

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

Throughout the inspection period, the inspectors reviewed weekly and daily work schedules to determine when risk-significant activities were scheduled. The inspectors discussed six selected activities with operations and work control personnel regarding risk evaluations and overall plant configuration control. The inspectors discussed emergent work issues with work control center personnel and reviewed the prioritization of scheduled activities. The inspectors verified the performance of plant risk assessments related to planned and emergent maintenance activities as required by 10 CFR 50.65(a)(4) and plant Procedure 01-S-18-6, "Risk Assessment of Maintenance Activities," Revision 1. Specific maintenance items reviewed during this period included:

- MAI 319291, Reactor core isolation cooling system turbine
- MAI 323209, Standby service water system flow indication (1C61R001B)
- MAI 323382, Main Steam Line C outboard isolation valve (1B21FO28C)
- MAI 323757, Division I emergency diesel generator voltage regulator
- MAI 323511, Division II emergency diesel generator voltage regulator
- MAI 325165, Residual heat removal Pump C

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Events (71111.14)

.1 Manual Scram of Control Rod 40-45

a. Inspection Scope

On December 3, 2002, the inspector observed GGNS perform a planned nonroutine reactivity adjustment by manually scramming Control Rod 40-45 in order to suppress the reactor core neutron flux around a suspected leaking fuel pin. The inspector observed control room shift personnel performing the pre-evolution brief, establishing prerequisites, manually scramming the control rod, operator procedural compliance and response for the evolution, and that the expected results were obtained.

b. Findings

No findings of significance were identified.

.2 Reactor Downpower Evolution to Facilitate Reactor Feed Pump Maintenance

a. Inspection Scope

On October 19, 2002, the inspectors observed operations personnel perform a planned nonroutine plant downpower from 100 percent to 50 percent rated thermal power to facilitate troubleshooting on the turbine driven reactor feed Pump B trip circuitry. The inspectors observed control room shift personnel performing the pre-evolution brief, establishing prerequisites, lowering reactor recirculation flow, manually inserting control rods, operator procedural compliance and response for the evolution, and that the expected results were obtained.

b. Findings

No findings of significance were identified.

.3 Nonroutine Downpower Evolution to Facilitate Main Steam Line Isolation Valve Maintenance

On November 7, 2002, the inspectors observed operations personnel perform a planned nonroutine plant downpower to 68 percent rated thermal power in order to allow troubleshooting on the Main Steam Line C outboard isolation valve. The inspectors observed control room shift personnel performing the pre-evolution brief, establishing prerequisites, lowering reactor recirculation flow, manually inserting control rods, operator procedural compliance and response for the evolution, and that the expected results were obtained.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors selected five operability evaluations conducted by GGNS personnel during the report period involving risk-significant SSCs. The inspectors evaluated the technical adequacy of the operability determinations, determined whether appropriate compensatory measures were implemented, and determined whether GGNS personnel considered all other pre-existing conditions, as applicable. Additionally, the inspectors evaluated the adequacy of the GGNS's problem identification and resolution program as it applied to operability evaluations. Specific operability evaluations reviewed are listed below.

- CR-GGN-2002-1810, Fuel channel bowing
- CR-GGN-2002-1970, Rod position indication operating experience
- CR-GGN-2002-2038, Safety relief valve environmentally qualified connectors
- CR-GGN-2002-2057, Division II emergency diesel generator
- CR-GGN-2002-2609, Standby service water piping degradation

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

The inspectors evaluated the cumulative effects of all the plant's significant operator workarounds for the following attributes: (1) the reliability, availability, and potential for misoperation of safety-related systems; (2) the ability of the operators to respond in a correct and timely manner to plant transients and accidents; and, (3) the potential for increasing an initiating event frequency or affecting multiple mitigating systems.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed postmaintenance test procedures and associated testing activities for four selected risk-significant mitigating systems. In each case, the associated work orders and test procedures were reviewed against the attributes in Inspection Procedure 71111, Attachment 19, to determine the scope of the maintenance activity and determine if the testing was adequate to verify equipment operability. The reviewed activities were:

- MAI 301746, Reactor water cleanup Pump B
- MAI 322526, Offgas post-treatment radiation Monitor B
- MAI 322993, Standby service water remote shutdown panel flow indication
- MAI 318980, Control HVAC Train B

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope

The inspectors observed licensee refueling outage planning and execution activities. The inspectors' review included scheduling, training, outage configuration management, decay heat removal operation and management, reactivity controls, inventory controls, tag out and clearance activities, foreign material exclusion management, and fuel movement and storage. Specific activities observed included:

- Drywell closeout inspections and containment integrity
- Reactor plant heatup and Mode 3 operations
- Reactor start up and Mode 2 operations
- Reactor power ascension and Mode 1 operations

b. Findings

Introduction

The inspectors identified a Green violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," when licensee maintenance personnel improperly restored the reactor recirculation Loop B decontamination flange to a leak tight configuration due to a lack of instructions for the task.

Description

On September 28, 2002, licensee personnel were refilling reactor recirculation Loop B in order to perform a valve stroke to retest the reactor recirculation Loop B flow control Valve B33F060B after maintenance on the valve. Following the refilling, GGNS mechanical maintenance personnel noted approximately 60 gpm leaking from the reactor recirculation loop decontamination flange and decided to tighten this flange in an effort to lower the leakage rate.

The flange was in place but its bolts had been loosened earlier to allow draining of the water around B33F060B for valve repacking maintenance. Maintenance personnel decided to tighten the flange just enough to lower the leakage rate, but not to fully torque the flange so that it could be easily removed for future maintenance prior to reactor startup. A full torquing sequence was the only method of flange installation detailed in the Maintenance Action Item (MAI) 303827 instructing the work. This interim tightening activity was not controlled by procedure even though it had the potential to affect the quality of the RCS pressure boundary. Also, licensee maintenance supervision did not specify how to accomplish the interim torquing in the prejob brief for the evolution.

Maintenance personnel then proceeded to the drywell to tighten up the flange to lower the leak rate from the flange. The inspector observed maintenance personnel tightening the flange fasteners without a proper torque wrench. In the absence of any prescribed instructions, the maintenance personnel used a slugging wrench and a four pound shop hammer on the flange bolting. This practice exerted uncontrolled, unknown, and possible excessive amounts of torque to the flange bolting. This could compromise their ability to withstand RCS pressure or any spikes in pressure from plant transients. Upon analyzing the torquing process after the inspector raised the issue with licensee supervision, the licensee concluded that torque applied to the bolts could not be quantified and replaced the bolting before the RCS was pressurized.

Analysis

This finding is more than minor because the objective of the initiating events cornerstone to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations was affected in the form of a loss of coolant from the flanged pressure boundary. Using Phase I of the SDP, the inspectors determined that the finding did not increase the likelihood of a fire or flooding and characterized the finding as Green or of very low safety significance. In this determination, the inspectors assumed that the bolts were overtorqued leading to a degraded RCS boundary at the decontamination flange and that the bolts would have remained in place if the inspectors had not brought this issue to the attention of GGNS supervision.

Enforcement

The inspectors determined that the failure of licensee personnel to prescribe adequate instructions for torquing the reactor recirculation Loop B decontamination flange which resulted in a degraded condition of an RCS pressure boundary was a violation of 10 CFR 50, Appendix B, Criterion V, which states in part, that activities affecting quality shall be prescribed by documented instructions of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions. However, this violation is being treated as an NCV (NCV 05000416/2002-005-01) because of the very low safety significance of this condition and because the licensee included this condition in their corrective action program in Condition Report CR-GGN-2002-1988. This condition report documents GGNS personnel’s evaluation which led to replacement of the bolts and procedural changes to prevent recurrence of this practice.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed performance of surveillance test procedures and reviewed test data of five selected risk-significant SSCs to assess whether the SSCs satisfied the Technical Specifications, the Updated Final Safety Analysis Report, the Technical Requirements Manual, and licensee procedural requirements; and, to determine if the testing appropriately demonstrated that the SSCs were operationally ready and capable of performing their intended safety functions. The following tests were inspected:

- 06-IC-1E31-Q-1002, "Main Steam Line Tunnel, RCIC Equipment Room High Temperature (PCIS) (RCIC ISOL) (RWCU ISOL)," Revision 102
- 06-OP-1C41-R-0002, "Standby Liquid Control Injection Test," Revision 108
- 06-OP-1E22-Q-0005, "High Pressure Core Spray Quarterly Valve Test," Revision 105
- 06-OP-1P75-R-0004, "Division II Emergency Diesel Generator 18 Month Functional," Revision 107
- 06-OP-1R20-W-009, "Plant AC and DC Electrical Power Distribution Weekly Lineup," Revision 104

b. Findings

No findings of significance were identified.

Emergency Preparedness [EP]

1EP2 Alert Notification System Testing (71114.02)

a. Inspection Scope

The licensee's siren testing program was compared with the guidance of NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1, and Federal Emergency Management Agency Document REP-10, "Guide for the Evaluation of Alert and Notification Systems for Nuclear Power Plant." The inspectors reviewed siren failure trend data and records for calendar year 2001 through the third quarter of calendar year 2002. The inspectors also reviewed Procedure10-S-01-12, "Radiological Assessment and Protective Action Recommendations," Revision 30.

b. Findings

No findings of significance were identified.

1EP3 Emergency Response Organization Augmentation Testing (71114.03)

a. Inspection Scope

The inspectors discussed with the licensee changes made in the installed systems and testing programs for automatic phone dialing systems and paging systems during calendar years 2001 and 2002, to evaluate the licensee's continued ability to staff emergency response facilities in accordance with the licensee emergency plan and the requirements of 10 CFR Part 50, Appendix E. The inspectors reviewed the results of the annual group pager test conducted October 28, 2002, and the following procedures:

- TQ-110, "Emergency Preparedness Training Program," Revision 1
- 1-S-04-21, "Emergency Preparedness Training Program," Revision 108
- 10-S-02-2, "Maintaining the VIP 2000," Revision 7
- 10-S-01-6, "Notification of Off-Site Agencies and Plant On-Call Emergency Personnel," Revision 36

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed Revisions 46, 47, and 48 to the Grand Gulf Emergency Plan to determine if the revisions decreased the effectiveness of the emergency plan.

b. Findings

No findings of significance were identified.

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies (71114.05)

a. Inspection Scope

The inspectors reviewed the following documents related to the licensee's corrective action program to determine the licensee's ability to identify and correct problems in accordance with the requirements of 10 CFR 50.47(b)(14) and 10 CFR Part 50, Appendix E:

- Manager Self-Assessment, "Benchmark-Fermi 2 Nuclear Power Plant," for off-hour drills and Technical Support Center staffing, May 6-10, 2002
- Peer Group Assessment Report, "Graded Exercise of March 6-7, 2002"

- Emergency Preparedness Assessment Report, November 4-7, 2002
- Quality Assurance Audit Report, QA-7-2002-GGNS-1, “Emergency Preparedness Program”
- Procedure 1-S-10-3, “Emergency Preparedness Department Responsibilities,” Revision 9
- Root Cause Evaluation Report 02-15, “EOF Diesel Fail to Start”
- Summaries of corrective action documents assigned to the emergency preparedness department between November 2001 and November 2002
- Details of Condition Reports: 2000-0149, -0805, -0922, 2001-0297, -1310, -1509, and -1571, 2002-0203, -0691, -0694, -0812, -0906, -0958, and -1686

b. Findings

No findings of significance were identified.

1EP6 Drill Observation (71114.06)

a. Inspection Scope

On November 19, 2002, the inspectors observed a planned licensee emergency preparedness quarterly drill. The inspectors reviewed the drill scenario to determine if it reflected realistic plant configurations. The inspectors observed GGNS personnel at various locations during the exercise including the control room simulator, the Technical Support Center, the Emergency Operations Facility, and the Operations Support Center. The inspectors primarily focused on the ability of the emergency response organization to properly classify the simulated emergency through recognition of emergency action levels, their ability to activate the station emergency plan and procedures, and their ability to make proper and timely notifications as appropriate.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY
Cornerstone: Occupational Radiation Safety [OS]

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope

The inspectors interviewed radiation workers and radiation protection personnel involved in high dose rate and high exposure jobs during Refueling Outage 12 activities. The inspectors also conducted plant walkdowns within the controlled access area and conducted independent radiation surveys of selected work areas. The following items were reviewed and compared with regulatory requirements:

- Quality Assurance Surveillance Reports QS-2001-GGNS-009, 010, and QS-2002-GGNS-011, 014, and 015

- Radiation Protection Self-Assessment, “Access to Radiologically Significant Areas,” documented in Condition Report LO-GLO-2002-0132

- Area posting and other controls for airborne radioactivity areas, radiation areas, high radiation areas, locked high radiation areas, and very high radiation areas

- Radiation work permits and radiological surveys involving airborne radioactivity areas and high radiation areas

- Access controls, radiological surveys, and radiation work permits for the following four significant high dose work jobs: Reactor Vessel Disassembly and Reassembly (2002-1403), Under Vessel Work (2002-1508), ISI/NDE Inside the Annulus and Drywell (2002-1516), and Diving in the Suppression Pool (2002-1528)

- Dosimetry placement when work involved a significant dose gradient

- Controls involved with the storage of highly radioactive items in the spent fuel pool

- A summary of access controls and high radiation area work practice related corrective action documents written since May 2001 and selected specific examples: (2002-1003, 2002-1143, 2002-1301, 2002-1504, 2002-1778, 2002-1809, 2002-1816, 2002-1820, 2002-1828, and 2002-1834)

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

The inspectors interviewed radiation workers and radiation protection personnel involved in high dose rate and high exposure jobs in the controlled access areas during normal operations. Field observations of selected work areas within the controlled access areas were conducted. The following items were reviewed and compared with regulatory requirements to determine whether the licensee had an adequate program to maintain occupational exposure as low as is reasonably achievable (ALARA):

- ALARA program procedures

- Processes used to estimate and track exposures
- Plant collective exposure history for the past 3 years, current exposure trends, and 3-year rolling average dose information
- Six radiation work permit (RWP) packages (2002-1004, 1012, 1403, 1505, 1523, and 1910) for work activities with the highest personnel collective exposures during the inspection period
- One job (RWP 2002-1080, “Remove and Replace TIP C”) was observed and tours were conducted in various areas of the turbine, auxiliary, and containment buildings
- Use of engineering controls to achieve dose reductions were evaluated for the RWP packages reviewed
- Exposures of selected work groups (radiation protection, operations, maintenance support, and mechanical maintenance)
- Hot spot tracking and reduction program
- Plant-related source term data, including source term control strategy
- Radiological work planning
- Four Quality Surveillance Reports (QS-2002-GGNS-006, 011, 014, and 015) and three Self-Assessments (January 21, 2002-“ALARA Planning and Controls,” July 31, 2002-“RP ALARA Outage Prep,” and August 28, 2002- “RP ALARA Planning and Control Assessment”)
- ALARA Committee meeting minutes conducted January 2002 through November 2002
- Selected corrective action documents involving the ALARA program and radiation worker practice deficiencies (Condition Reports: CR-GGN-2002-00387, 00482, 00736, 01002, 01156, 01160, 01162, 01401, 01453, 01509, 01521, 01638, 01874, and 01895)
- Declared pregnant worker dose monitoring controls

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator Verification (71151)

.1 Safety System Unavailability

a. Inspection Scope

The inspectors verified the accuracy and completeness of the data used to calculate and report performance indicator information for two indicators from the fourth calendar quarter 2001 through the third calendar quarter 2002. The inspectors used Nuclear Energy Institute 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 2, as guidance and interviewed licensee personnel responsible for compiling the information.

- Emergency AC power system unavailability
- Residual heat removal system unavailability

b. Findings

No findings of significance were identified.

.2 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspectors reviewed corrective action program records for technical specification required locked high radiation areas, very high radiation areas (as defined in 10 CFR 20.1003), and unplanned exposure occurrences (as defined in NEI 99-02) for the past 12 months to confirm that these occurrences were properly recorded as performance indicators. Controlled access area entries with exposures greater than 100 millirem were reviewed, and selected examples were examined to determine whether they were within the dose projections of the governing radiation work permits. Whole-body counts or dose estimates were reviewed if the radiation worker received a committed effective dose equivalent of more than 100 millirem.

The inspectors also performed routine checks, while on tours throughout the plant, to ensure that locked high radiation areas were properly secured.

b. Findings

No findings of significance were identified

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

a. Inspection Scope

The inspectors reviewed radiological effluent release program corrective action records, licensee event reports, and annual effluent release reports documented during the past four quarters to determine if any doses resulting from effluent releases exceeded the performance indicator thresholds (as defined in NEI 99-02).

b. Findings

No findings of significance were identified.

.4 Emergency Response Organization Drill Participation

a. Inspection Scope

The inspectors reviewed the following records related to emergency response organization participation in order to verify the licensee's reported data:

- Emergency response organization rosters for the first three quarters of calendar year 2002

- List of key emergency response organization positions

- Drill participation records for a sample of eight key responders for drills conducted during the first three quarters of calendar year 2002

- Performance indicator summary sheets and reports

b. Findings

No findings of significance were identified.

.5 Alert and Notification System

a. Inspection Scope

The inspectors reviewed siren testing records for a 100 percent sample of tests conducted for the first three quarters of calendar year 2002, to verify the accuracy of data reported for this performance indicator.

b. Findings

No findings of significance were identified.

.6 Drill and Exercise Performance

a. Inspection Scope

The inspectors reviewed the following documents related to the drill and exercise performance indicator in order to verify the licensee's reported data:

- Procedure LI-107, "NRC Performance Indicator Process," Revision 1
- Procedure 10-S-04-4, "Performance Indicators," Revision 3
- Drill schedules for calendar year 2002
- Drill scenarios, notification forms, and participant logs for drills conducted during the first three quarters of calendar year 2002
- Drill evaluation records for all drills conducted during the first three quarters of calendar year 2002
- Performance indicator summary sheets and reports

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

a. Inspection Scope

The inspectors assessed the licensee's problem identification and resolution efforts associated with recent small bore piping fatigue failures on Train A of the RHR system. The inspectors reviewed the licensee's evaluations for operability and reportability of the issue, verified corrective actions were appropriately focused to correct the problem, determined that those corrective actions were completed in a manner commensurate with the safety significance of the issue, and whether a proper extent of condition was determined. The assessment also included a review of previously identified deficiencies associated with RHR system vibration.

b. Findings

Two very low safety-significant findings were identified following the small bore piping failures of the RHR Train A system.

Self-Revealing Event Summary

On September 16, 2002, GGNS was conducting a refueling outage with the reactor in Mode 5 of operation and RHR system Train A in shutdown cooling when they experienced a lowering reactor vessel water level, lowering suppression pool level, and a

“HI-HI RHR Room Sump” level alarm followed by a “RHR Room Flood” alarm. Operators entered emergency procedure 05-S-01-EP4 “Auxiliary Building Control.” Investigation revealed that the RHR system had experienced two small-bore piping fatigue failures on the in-service RHR Train A shutdown cooling train due to severe vibration while operating the heat exchanger outlet Valve (E12-F003A) beyond its optimum throttling range.

During the event GGNS personnel found approximately 8"-12” of water on the room floor and identified two leak locations. One piping failure had occurred in the pressure-locking bypass piping on Valve E12-F024A (RHR system test return to suppression pool) resulting in the loss of some suppression pool inventory into the RHR Train A room. The second failure had occurred at the RHR heat exchanger conductivity cell isolation Valve (E12-FX-060) and resulted in the loss of some reactor vessel inventory into the RHR Train A room. The GGNS operating crew turned off RHR system Train A and installed temporary plugs at the two failure locations to stop the loss of suppression pool and RHR system piping inventory. The GGNS operating crew in the control room shifted shutdown cooling to the alternate decay heat removal (ADHR) system followed by returning RHR Train B to service. The failed small bore piping was later repaired and the system returned to service.

<u>Previous Corrective Actions</u>

Since 1984, GGNS has experienced severe flow induced vibration in the RHR system resulting from throttling system flow with the E12-F003 Valves partially open 15 percent or less. The following examples were missed opportunities for GGNS to fully understand, identify, and correct a condition which resulted in component failures and eventually rendered the system inoperable:

- On October 17, 1984, Licensee Event Report (LER) 84-24-2 reported a forced shutdown caused by both RHR trains inoperable due to discovered piping support deficiencies following the identification of cracks found on the RHR Loop B. The root cause of the pipe cracking was determined to be from abnormal system vibrations attributed to throttling the F003 Valve less than 15 percent open with the RHR heat exchanger bypass Valve (F048) closed. The licensee revised the RHR system operating instruction to prevent throttling both the F003 and F048 Valves at the same time but failed to identify that throttling full flow through the F003 Valve was beyond its design capability.

- On April 13, 1989, during a refueling outage, the F003B Valve failed to fully stroke close when establishing conditions for a surveillance test as documented in Condition Report 1989-0146. The valve was disassembled for inspection and found to be damaged from apparent excessive vibration. The valve was removed and replaced. This was a second opportunity that the licensee failed to identify that throttling full flow through the F003 Valve was beyond its design capability and take the appropriate corrective actions to prevent recurrence.

- On November 15, 1993, MAI 111603 was written to reinstall the valve handwheel and replace or tighten missing and loosened screws on the E12-F003A Valve actuator assembly resulting from excessive vibration of the system while throttling flow while in shutdown cooling. The licensee did not perform a causal analysis or take appropriate corrective action to preclude or prevent future system vibration and component failures.

- On November 16, 1999, GGNS again experienced severe vibration of RHR Train A while throttling F003A and F048A as documented in Condition Report 1999-1706. GGNS determined that the apparent cause of the severe vibration was not previously addressing the inherent design limitations for throttling RHR shutdown cooling flow with E12-F003A(B) but failed again to take appropriate corrective action to preclude or prevent future system vibration and component failures.

10 CFR Part 50, Appendix B, Criterion XVI, requires, in part, that measures be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. For significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective actions taken to preclude recurrence. The identification of significant conditions adverse to quality shall be documented and reported to the appropriate levels of management.

The inspectors determined that several previous RHR system method of operation deficiencies were not properly evaluated and corrected by the licensee to prevent recurrence. This finding was considered a Green noncited violation (NCV 05000416/2002-005-02) of 10 CFR Part 50, Appendix B, criterion XVI, "Corrective Actions." Each previous deficiency involved a missed opportunity to fully understand and correct a significant condition adverse to quality associated with operating the RHR system heat exchanger outlet Valves E12-F003A(B) outside of their optimum throttling range resulting in severe system vibration and ultimately component damage rendering the system inoperable.

<u>Valve E12-F024A Pressure Locking Modification Design Deficiency</u>

GGNS Root Cause Determination Report CR-GGN-2002-1779 identified a deficiency in the design analysis used in the pressure locking design modification to Valve E12-F024A. In June 1996, GGNS modified RHR Valve E12-F024A to prevent pressure locking of the valve's internals by installing a new bypass piping configuration. The design modification did not take into account the calculated vibration frequency of the pressure locking piping. The licensee failed to adhere to GGNS Design Document M-18, "Engineering Users Manual for Routing and Supporting Two Inch and Under Piping." This document required cantilever vent and drain pipes with a computed frequency of 33 Hz or less be made more rigid by means of shortening the length of the cantilever to increase its frequency or installing additional piping supports. The licensee did neither. As a result, this design condition created a resonance and low stress high-cycle fatigue in the piping whenever the pump was run causing a fatigue crack at a socket weld location which ultimately contributed to the fracture of a socket weld connecting the pressure locking piping to the RHR Train A piping header during the event on September 16, 2002.

10 CFR Part 50, Appendix B, Criterion III, requires, in part, that design control measures shall provide for verifying or checking the adequacy of design by the performance of design reviews which shall be subject to design control measures commensurate with those applied to the original design. Original GGNS Design Document M-18, "Engineering Users Manual for Routing and Supporting Two Inch and Under Piping," provided design guidance that was not complied with by the licensee, which remained undetected until the small bore piping failures occurred. The guidance was not complied with by GGNS and the independent review of the piping modification did not identify this deficiency. This finding was considered a Green noncited violation (NCV 05000416/2002-005-03) of 10 CFR Part 50, Appendix B, criterion III, "Design Control."

These findings were individually evaluated in accordance with NRC Inspection Manual Chapter 0612. These findings affected the mitigating systems cornerstone of the reactor oversight process and were considered more than minor because they were viewed as precursors to a significant event resulting in a loss of the RHR system Train A. The inspectors utilized Inspection Manual 609, Appendix G, “Shutdown Operations Significance Determination Process,” to determine the significance of this finding. The risk significance of this finding was determined to be very low. The significance determination process assumed that both emergency diesels generators, high pressure core spray, low pressure core spray, and standby service water remained available. No credit was given to RHR B since it was out of service for maintenance. Credit was given to the fire water system for makeup to the reactor vessel since the vessel head was removed and a fire hose could be used. These conditions were documented in the licensee's correction action program as CR-GGN-2002-1779.

4OA6 <u>Meetings, including Exit</u>

The inspectors presented the results of the access control to radiologically significant areas inspection to Mr. J. Edwards, General Manager, and other members of licensee management at the conclusion of the inspection on October 3, 2002.

The inspector presented the ALARA inspection results to Mr. J. Roberts, Director of Nuclear Safety Assurance, and other members of licensee management at an exit meeting on November 22, 2002.

The inspector presented the Emergency Preparedness inspection results to Mr. J. Roberts and other members of licensee management at an exit meeting on December 12, 2002.

On January 8, 2003, the resident inspectors presented the inspection results to Mr. J. Roberts and other members of licensee management.

The inspectors also asked if any materials examined during the inspections should be considered proprietary. No proprietary information was identified by the licensee.

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 an NCV.

Technical Specification 5.4.1.a requires in part that written procedures be implemented covering the activities in Regulatory Guide 1.33, “Quality Assurance Program Requirements,” Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Revision 2, Appendix A, Section 7.e(1) references radiation protection procedures for access control to radiation areas including a RWP system. Procedure 01-S-08-34, Revision 4, Section 6.2.4.a.3, requires that each person entering a radiologically posted area or any other area requiring an RWP must read, understand, and obey the terms and conditions of the RWP. On February 28, June 7, June 27, August 7, August 15, August 30, and September 21, 2002, individuals entered the controlled access area using the wrong RWPs. These instances are described in the licensee’s corrective action program Condition Reports CR-GGN-2002-00387, 01002, 01162, 01453, 01509, 01638, and 01874. Because it did not involve ALARA planning and controls, there was no personnel overexposure, there was no substantial potential for personnel overexposure, and the finding did not compromise the licensee’s ability to assess dose, this violation is not more than of very low significance, and is being treated as a noncited violation.

ATTACHMENT

PARTIAL LIST OF PERSONS CONTACTED

Licensee

C. Abbott, Quality Assurance Supervisor
D. Barfield, Manager, System Engineering
R. Barnes, Manager, Training and Development
R. Benson, Supervisor, Radiation Protection
C. Bottemiller, Manager, Plant Licensing
K. Christian, Superintendent, Mechanical Maintenance
W. Eaton, Vice President, Operations
N. Edney, Supervisor, Radiation Protection
J. Edwards, General Manager, Plant Operations
C. Ellsaesser, Manager, Corrective Action and Assessment
M. Guynn, Manager, Emergency Preparedness
M. Larson, Senior Licensing Specialist
R. Moomaw, Manager, Outage Planning and Scheduling
J. Roberts, Director, Nuclear Safety Assurance
J. Robertson, Manager, Quality Assurance
M. Rohrer, Manager, Maintenance
F. Rosser, Supervisor, Radiation Protection
G. Sparks, Manager, Operations
D. Wiles, Director, Engineering
R. Wilson, Superintendent, Radiation Protection
H. Yeldell, Manager, Design Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000416/2002-005-01	NCV	Failure to prescribe instructions for tightening a reactor recirculation system flange allows unquantifiable torquing of bolts which construct part of the reactor coolant system boundary (Section 1R20)
05000416/2002-005-02	NCV	Inadequate corrective actions associated with operating the residual heat removal system heat exchanger outlet Valve (E12-FO-3A) beyond its optimum throttling range leads to excessive system vibration and small bore piping failures (Section 4OA2)
05000416/2002-005-03	NCV	Inadequate design controls associated with adding a permanent pressure locking modification to a RHR system valve resulted in a resonance and low stress high cycle fatigue whenever the RHR pump was run which ultimately contributed to the fracture of a

socket weld connecting the pressure locking piping to the RHR Train A piping (Section 4OA2)

LIST OF DOCUMENTS REVIEWED

Procedures:

Grand Gulf Nuclear Station Unit 1 Fire Preplans, Volumes 1 and 2, Revision 11

Condition Reports:

2002-01952	2002-02246	2002-02256	2002-02364
2002-01996	2002-02247	2002-02257	2002-02373
2002-01997	2002-02248	2002-02259	2002-02384
2002-02001	2002-02251	2002-02265	2002-02623
2002-02151	2002-02253	2002-02269	
2002-02235	2002-02254	2002-02300	
2002-02243	2002-02255	2002-02361	

Maintenance Action Items:

301518	321090	324159	315512
309087	321248	325560	320568
314366	321689	325563	314366
318995	323125	325568	321792
319919	323738	325575	318102
321089	324157	325704	

Other Miscellaneous Documents:

Engineering Standard ES-19, "Office and Field Engineering Manual for Routing and Supporting Two Inch and Under Piping," Revision 20

Grand Gulf Nuclear Station Operations Burden List dated December 26, 2002