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

July 27, 2001

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 INTEGRATED INSPECTION REPORT 50-416/01-03

Dear Mr. Eaton:

On June 30, 2001, the NRC completed an inspection at your Grand Gulf Nuclear Station. The enclosed report documents the inspection findings which were discussed on July 5, 2001, with Mr. Joe Venable and other members of your staff.

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

Based on the results of this inspection, the inspectors identified five violations of NRC requirements. These issues were evaluated under the risk significance determination process as having very low safety significance. Because they have been entered into your corrective action program, the NRC is treating these violations as noncited violations (NCVs), in accordance with Section VI.A of the Enforcement Policy. If you deny these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the 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/NRC/ADAMS/index.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/

Jeffrey A. Clark, Chief
Project Branch A
Division of Reactor Projects

Docket: 50-416
License: NPF-29

Enclosure:
NRC Inspection Report
50-416/01-03

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

Sam Mabry, 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. F. E. Thompson, Jr.
State Health Officer
State Board of Health
P.O. Box 1700
Jackson, Mississippi 39205

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 from ADAMS by RIV:
Regional Administrator **(EWM)**
DRP Director **(KEB)**
DRS Director **(ATH)**
Senior Resident Inspector **(TLH4)**
Branch Chief, DRP/A **(JAC)**
Senior Project Engineer, DRP/A **(DNG)**
Staff Chief, DRP/TSS **(PHH)**
RITS Coordinator **(NBH)**
Scott Morris **(SAM1)**
NRR Event Tracking System **(IPAS)**
GG Site Secretary **(MJS)**
Dale Thatcher **(DFT)**

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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-416
License:	NPF-29
Report:	50-416/01-03
Licensee:	Entergy Operations, Inc.
Facility:	Grand Gulf Nuclear Station
Location:	Waterloo Road Port Gibson, Mississippi 39150
Dates:	April 1 through June 30, 2001
Inspectors :	T. Hoeg, Senior Resident Inspector P. Alter, Resident Inspector D. Carter, Health Physicist J. Dodson, Health Physicist J. Whittemore, Senior Reactor Inspector C. Johnson, Senior Reactor Inspector B. Nicholas, Senior Health Physicist
Approved By:	Jeffrey A. Clark, Chief, Project Branch A

SUMMARY OF FINDINGS

IR 05000416-01-03, on 04/01-06/30/2001, Entergy Operations, Inc, Grand Gulf Nuclear Station. Integrated res & reg insp rpt; Rad Access Control; ALARA Planning & Controls; Rad Mat Processing & Shipping; Inservice Insp Activities; Heat Exchanger Perf

The inspection was conducted by resident inspectors, regional health physicists, and regional reactor inspectors. The inspection identified five Green noncited violations. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using IMC 609 "Significance Determination Process" (SDP). Findings for which the SDP does not apply are indicated by "No Color" or by the severity level of the applicable violation. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html.

A. Inspector Identified Findings

Cornerstone: Mitigating Systems

- Green. The inspectors determined that the licensee failed to perform adequate corrective actions to replace a control logic relay, for the Division II emergency diesel generator building ventilation system, which was previously identified as being susceptible to failure and required replacement. Failure to replace the subject relay, following previous identified failures, constituted inadequate corrective action and is a violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action." This violation is noncited in accordance with Section VI.A of NRC's Enforcement Policy and is in the licensee's corrective action program (CR-GGN-2001-1072). This finding also had crosscutting aspects in the area of problem identification and resolution.

  This finding was of very low safety significance because, although the Division II emergency diesel generator (EDG) outside air fan would not have automatically started in fast speed, the diesel was able to perform it's safety function with the fan in slow speed and room temperature below 120 degrees F and because the operators would still have the opportunity to manually shift the fan to fast speed prior to the room reaching 120°F (Section 4OA2).

- Green. The licensee's program for evaluating the thermal performance of safety-related heat exchangers identified degraded performance of the risk-significant, safety-related high pressure core spray pump room cooler. The degraded condition of the room cooler occurred between December 12, 1997, and May 22, 2000, for an approximate 30-month period. The condition was eventually determined to be caused by an improper technique for collecting the cooler air flow data, which rendered the cooler capacity software calculation unreliable. The failure to promptly correct the room cooler test deficiencies was a violation of Criterion XVI of Appendix B to 10 CFR Part 50. This violation is being treated as a noncited violation, consistent with Section VI. A of the NRC Enforcement Policy. This violation (50-416/0103-01, Section 1R07) has been entered into the licensee's corrective action program in Condition Report CR-GGN-2001-0591. This finding also had crosscutting aspects in the area of problem identification and resolution.

The finding that the high pressure core spray room cooler was degraded was of very low safety significance because all mitigation systems remained operable and barrier integrity was not challenged. Following the inspection, the licensee entered the finding into the corrective action program in Condition Report CR-GGN-2001-0591 (Section 1R07).

Cornerstone: Occupational Radiation Safety

- Green. An inspector determined that radiation levels in the Residual Heat Removal Pump A room were significantly higher than posted levels due to recent changes in plant operating conditions. The failure to perform a radiological survey is a violation of 10 CFR 20.1501(a). This violation is being treated as a noncited violation consistent with Section VI. A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as CR-GGN-2001-0674.

  The safety significance of the finding was determined to be very low by the Occupational Radiation Safety SDP because there was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised. The violation was more than minor because the failure to perform a radiological survey has a credible impact on safety and the potential for unplanned or unintended dose (Section 2OS1).

- Green. On June 6, 2001, the inspector determined that the contamination levels on top of the drywell head were significantly higher than the contamination levels measured around the flange area of the drywell head. The drywell flange area survey results were used as the radiological conditions for the top of the drywell head. These conditions were also used to brief a worker and prescribe protective clothing and respiratory protection prior to assigning work on top of the drywell head. As a result, a worker became contaminated and was assigned an internal dose of 10 millirem. No survey of the top of the drywell head was performed until after the contamination event. The failure to evaluate the concentrations or quantities of radioactive material and the potential radiological hazards on top of the drywell head prior to assigning work in that area is a violation of 10 CFR 20.1501(a). This violation is being treated as a noncited violation consistent with Section VI. A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as Condition Report CR-GGN-2001-1088.

  The safety significance of this violation was determined to be very low by the Occupational Radiation Safety SDP because there was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised. The violation was more than minor because the failure to perform a radiological survey resulted in an unintended dose (Section 2OS2).

Cornerstone: Public Radiation Safety

- Green. The inspector identified that the licensee excluded measured Pu-241 analysis results from the Reactor Water Cleanup (RWCU) A resin waste stream scaling factors on February 9, 2001. Radioactive Waste Shipment 2001-0203, containing RWCU-A resin, was classified using those scaling factors, manifested, and shipped on

February 12, 2001, without determining or documenting the Pu-241 activity. The licensee confirmed that no other shipments during the inspection period were affected. Because the licensee excluded measured Pu-241 activity from their scaling factors, reasonable assurance was not provided that the indirect method of identifying radionuclides in that waste stream was valid. Therefore, the exclusion of Pu-241 in the waste classification and manifest for Radioactive Waste Shipment 2001-0203 was a violation of 10 CFR Part 20, Appendix G. This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as Condition Report 2001-0994.

The safety significance of this violation was determined to be very low by the Public Radiation Safety SDP because radiation limits were not exceeded and there was no breach of package during transit, certificate of compliance problem, low level burial ground access problem, or failure to make notifications or provide emergency information. The violation was more than minor because there was a credible impact on safety, and it involved an occurrence in the licensee's radioactive material transportation program (Section 2PS2).

B. Licensee Identified Violations

Three violations of very low safety significance were identified by the licensee and were reviewed by the inspectors. Corrective actions taken or planned by the licensee appear reasonable. These violations are listed in Section 4OA7 of this report.

Report Details

Summary of Plant Status: During this inspection period, the plant operated at 100 percent power until the reactor was shut down on April 13, 2001, for a planned refueling outage. The refueling outage was completed on May 5, 2001. The reactor was returned to 100 percent power, where it operated throughout the remainder of this inspection period.

1. REACTOR SAFETY

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

1R01 Adverse Weather (71111.01)

a. Scope

The inspectors reviewed Grand Gulf Nuclear Station's (GGNS) readiness to mitigate severe weather conditions such as hurricanes, tornadoes, and flooding. Procedures OS-1-02-VI-1, Revision 101, "Flooding," and OS-1-02-VI-2, Revision 103, "Hurricanes, Tornadoes, and Severe Weather," were reviewed and site walkdowns performed to verify that the standby service water system (ultimate heat sink) and site drainage canal were ready for adverse weather conditions. The inspection verified that the ultimate heat sink cooling towers and components were protected from high winds and flooding conditions, to preclude the loss of standby service water design function.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

The inspectors performed partial system walkdown inspections of systems, important to reactor safety, in order to verify the operability of the system while the alternate train was out of service for planned maintenance. The inspectors reviewed system operating instructions, system valve and breaker lineups, operator logs, and system control room indications. They also verified that valves, breakers, and control circuits were in their required positions for operability. The following systems were inspected:

- Division II standby diesel generator
- Residual heat removal system Train B
- High pressure core spray

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors reviewed associated area fire plans, and performed walkdowns of these areas, to assess the material condition and operational status of fire detection, suppression systems, and equipment. The inspectors also examined the material condition of fire barriers, and control of transient combustibles. Specific risk significant areas included:

- Division I Engineered Safety Features Switchgear Room 1A309
- Division II Engineered Safety Features Switchgear Room 1A207
- Division III Engineered Safety Features Switchgear Room 1OC210
- Division III Engineered Safety Features Battery Room 1OC209
- Auxiliary Building Steam Tunnel Room 1A305
- Control Rod Drive Hydraulic Pumps Area 1A117

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

The purpose of this effort was to examine the licensee’s program to assure that heat sink or heat exchanger performance problems with a potential to increase risk at the Grand Gulf Nuclear Station would be identified. A secondary purpose of this effort was to determine if the licensee’s program was capable of identifying conditions or deficiencies that could mask degraded performance of safety-related heat exchangers or heat sinks. The heat sinks and heat exchangers sampled for review were:

- Standby service water system cooling tower and basin (ultimate heat sink)
- Residual heat removal system heat exchangers
- High pressure core spray pump room cooler

.1 Performance of Testing, Maintenance, and Inspection Activities

a. Inspection Scope

The inspector held discussions with system engineers, design engineers, and supervisors to review the licensee's test methodology and procedures for the selected heat exchangers. The inspector also reviewed testing, inspection, and maintenance requirements, methods, and procedures for the standby service water pumps, cooling tower fans, basin, and fill. Finally, the inspector reviewed results of testing and performance inspections for the ultimate heat sink to determine if the currently required periodic performance of integrated inspection, maintenance, and testing provided assurance that the ultimate heat sink remained operable and would perform its safety function.

The inspector compared the design requirement for heat removal performance of Residual Heat Removal Divisions I and II with the design capacity of the Divisions I and II heat exchangers. The inspector also reviewed the procedures for heat exchanger thermal performance testing and the results of recent thermal performance testing. The same review and comparison were performed for the high pressure core spray pump room cooling heat exchanger. For these heat exchangers, the inspector verified that the heat exchanger thermal performance test procedures established appropriate test conditions. The inspector also verified that proper measuring and test instrumentation that adequately accounted for test instrument uncertainties were specified. Also, the inspector verified that the performance of the tests would produce a valid indication of heat exchanger thermal performance. The inspector also verified that the test results were trended, causes of degrading trends were identified, and necessary action, in the form of cleaning or maintenance, was taken when test performance limits were approached. The inspector verified that the licensee's program assured that the safety-related heat exchangers were adequately monitored and maintained to assure the on-demand performance of the safety function.

b. Findings

No findings of significance were identified.

.2 Verification of Conditions and Operations Consistent with Design Bases

a. Inspection Scope

The inspector verified that the heat sink and heat exchanger test acceptance criteria were consistent with the design bases. The inspector then reviewed selected heat sink and heat exchanger performance calculations. This review verified that the licensee's staff had correctly identified the performance conditions and parameters to be attained, and using conservative assumptions had validated the existence of adequate thermal margin performance for the 30-day postevent period. Through additional review of the appropriate procedures and guidance, the inspector attempted to verify that all assumed configuration changes and operator actions required to preserve the assumptions of the analysis were, in fact, required by approved procedures and guidance.

b. Findings

No findings of significance were identified.

.3 Identification and Resolution of Problems

a. Inspection Scope

The inspector used Inspection Procedure 71152 as a guide for reviewing corrective action issues. The inspector examined the licensee's corrective action program for any problems associated with the selected heat exchangers by reviewing condition reports that identified issues related to degradation or masking of heat exchanger performance. The inspector also examined work orders and maintenance action items in which the licensee's staff had documented problems that were considered below the threshold of the highest tier of the corrective action program. The inspector verified that the licensee's program could identify conditions related to single or common cause failures that could lead to degraded heat exchanger performance.

b. Findings

The inspector identified a violation of Criterion XVI of Appendix B to 10 CFR Part 50. The licensee failed to promptly correct a condition that resulted in degraded surveillance test results of the high pressure core spray pump room cooler heat exchanger.

The inspector noted the following performance testing schedule and results for the high pressure core spray pump room cooler heat exchanger.

DATE	TEST RESULTS	RESULTS DISPOSITION
April 12, 1996	Operable	Retest Next Fuel Cycle
December 12, 1997	Test Aborted	Retest On Future Date to Meet GL 89-13 Commitment to Test Every Fuel Cycle
May 2, 1998	Inoperable	Retest As Soon As Possible
May 2, 1998	Degraded	Maintenance Action Item 292129
May 8, 1998	Degraded	Maintenance Action Item 292129
May 8, 1998	Degraded	Maintenance Action Item 292129
February 6, 1999	Degraded	Maintenance Action Item 292129
May 22, 2000	Operable	Retest Next Fuel Cycle

Through discussions with licensee personnel the inspector determined that the cooler was considered degraded when it would not maintain the design room temperature of 150°F as referenced in the equipment qualification data sheets under the conditions assumed in the design basis accident analysis. According to Technical Requirements Manual 6.7.3, the cooler was considered operable if the temperature remained below 180°F during design basis accident conditions. However, the inspector noted that between 150°F and 180°F the facility would be operating in an action statement of Technical Requirements Manual 6.7.3, which required collection of data and analysis on the effects of operating at elevated room temperature. Therefore, the degraded condition meant that under assumed design basis accident conditions the equipment and components in the room would perform the safety functions if room temperature remained below 180°F.

The inspector requested that the licensee provide a list of safety-related heat exchangers with risk fractions. This list indicated that the high pressure core spray room cooler had the highest risk fraction of all safety-related heat exchangers with a risk reduction worth of 1.92 E-2 and a risk achievement worth of 10.343. Using the information in the table above, the inspector determined that the degraded performance of the room cooler occurred between December 12, 1997, and May 22, 2000, for an approximate 30-month period. The condition was eventually determined to be caused by an improper technique for collecting the cooler air flow data, which rendered the cooler capacity software calculation unreliable. The degraded condition of the high pressure core spray room cooler was never entered into the licensee's corrective action program. The licensee's management indicated their belief that the maintenance action item program was adequate to correct the issue. The inspector concluded, however, that the licensee failed to promptly correct the room cooler test deficiencies.

The inspector referenced Manual Chapter 610*, Appendix B, Group 1 Questions, and determined that untimely corrective action would become a more significant safety concern if associated with an actual degraded room cooler (Question 3); therefore, the issue was more than minor. In referencing the Group 2 Questions, the inspector determined that the loss of the high pressure core spray pump room cooler could credibly affect the function of a system or train in a mitigating system (Question 2). On the basis of these determinations, the inspector assessed the issue using the SDP.

This violation was of low safety significance (Green) because there was not an actual loss of safety function of a system, the Technical Specification allowed outage time was not exceeded for a single train as the result of an actual loss of safety function, there was not an actual loss of safety function for greater than 24 hours for equipment designated as risk-significant in accordance with 10 CFR 50.65, and this issue did not involve external events. Criterion XVI of Appendix B to 10 CFR Part 50 states "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected." Contrary to this, the inspector found the failure to promptly identify and correct the improper testing technique that resulted in the failed surveillance results for the risk-significant, safety-related high pressure core spray pump room cooler to be a violation of Criterion XVI of Appendix B to 10 CFR Part 50. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. This violation (50-416/0103-01) has been entered into the licensee's corrective action program as Condition Report CR-GGN-2001-0591.

1R08 <u>Inservice Inspection Activities (71111.08)</u>

.1 <u>Performance of Nondestructive Examination (NDE) Activities</u>

The Grand Gulf Nuclear Station inservice inspection (ISI) program was committed to the 1992 Edition with portions of the 1993 Addenda of ASME Section XI Code, except that ultrasonic examinations shall be in accordance with the 1977 Edition with the Summer of 1979 Addenda as specified in the ISI plan for the first interval. The licensee was currently in the third period of the second interval of the program.

a. Inspection Scope

The inspector requested and reviewed the licensee's NDE records for work that was performed for the current outage. The following NDE records were reviewed:

System	Component/Weld Identification	Examination Method
Reactor Coolant System	B13-N07-KA	Ultrasonic Examination (UT)
Reactor Coolant System	B13-N08-IR	UT
Feedwater System	B21G030DI00A-5	UT
Main Steam System	B21G116W15	Magnetic Particle (MT)
Feedwater System	B21G230W16	MT
Reactor Recirculation	B33G10-B1-F	UT, Dye Penetrant (PT)
Reactor Recirculation	B33G10-B1-H	UT, PT
Reactor Water Cleanup	G33G002W54	PT
Reactor Water Cleanup	G33G002W54	PT
Standby Liquid Control	C41G120W11	PT
Residual Heat Removal	E12G012W49	MT, UT
Reactor Core Isolation Cooling	E51G001W40	UT
Reactor Water Cleanup	G33G001W904	UT

The inspector reviewed licensee NDE and contractor personnel qualification and certification records to determine if the NDE personnel were certified to perform the above examinations and UT calibration records.

The inspector reviewed the implementation of the licensee's containment ISI program plan for inspection against the requirements of ASME Section XI, Subsection IWE.

b. Findings

No findings of significance were identified.

.2 ASME Code Repair and Replacement Activities

a. Inspection Scope

The inspector reviewed maintenance action items (MAIs) for ASME code repair and replacement work activities performed during the current outage. The inspector reviewed MAIs 275666 and 275668, modification to Main Steam Isolation Valves B21F028B and B21F028D, respectively; and MAIs 282771 and 286800, replacement of Containment Isolation Valves G33F039 and G33F040, respectively. This review included review of radiography film for Valves G33F039 and G33F040.

The inspector also reviewed a non-ASME code repair activity performed in accordance with ASME Code Case N-561, which was not included in Regulatory Guide 1.147 and had not been approved by the NRC at the time of this inspection. The inspector reviewed the licensee's proposed submittal to the Office of Nuclear Reactor Regulation for technical concerns. The inspector participated in a conference call between the licensee and the Office of Nuclear Reactor Regulation during this inspection, in which the Office of Nuclear Reactor Regulation gave verbal approval of the use of ASME Code Case N-561. The inspector then reviewed MAI 298457 that detailed the work activity for the performance of weld overlay on a pinhole leak downstream of Valve E12-F018A (Residual Heat Removal Loop A minimum flow manual isolation valve) at the 90 degree elbow.

b. Findings

No findings of significance were identified.

.3 Problem Identification and Resolution

a. Inspection Scope

The inspector performed a detailed review of a sample of condition reports initiated within the past 2 years in the area of ISI activities.

The review was conducted to ascertain that the licensee’s corrective action program was identifying performance issues within the ISI program. This review assessed the effectiveness of cause determination, corrective action, and the adequacy of the licensee's effort to identify transportability and generic issues. The review also assessed the effectiveness of the licensee's effort to identify and address programmatic issues within the ISI program.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

On June 11, 2001, the inspectors observed simulator training of an operating crew as part of the operator requalification training program. Emphasis was placed on observing weekly evaluation exercises of high risk licensed operator actions, operator activities associated with the emergency plan, and lessons learned from industry and plant experiences. In addition, the inspectors compared simulator control panel configurations with the actual control room panels for consistency, including recent modifications implemented in the plant.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

.1 Resident Baseline Inspection

a. Inspection Scope

The inspectors reviewed performance-based problems involving 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 system health reports and system functional failures for the last 2 years. The following SSCs were reviewed:

- Standby liquid control system
- Reactor water cleanup system
- Residual heat removal system
- Control rod drive hydraulic system
- Main steam isolation valves
- Emergency diesel generator building ventilation system

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 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 items reviewed during this period included:

- MAI 288763, Division I standby diesel generator planned outage maintenance
- MAI 268930, Division II standby diesel generator 10-year ISI maintenance
- MAI 297324, Bus 13AD maintenance outage
- MAI 296760, Division II standby service water cooling tower Fan C maintenance
- MAI 296017, Standby Liquid Control A pulsation dampener repair maintenance

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Plant Evolutions (71111.14)

.1 Draining of the Upper Containment Pool to the Suppression Pool

a. Inspection Scope

The inspector observed licensee operations personnel performing nonroutine planned outage draindown evolutions. The evolution observed was the draining of the upper containment pool to the suppression pool in preparation for reactor vessel disassembly and reassembly. The inspector reviewed operator logs, level instrumentation data, and Procedure 04-1-01-P11-2, "Refueling Water Storage and Transfer System," Revision 46.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed selected operability evaluations, affecting risk significant mitigating systems, to assess: (1) technical adequacy of the evaluations; (2) whether continued system operability was warranted; (3) whether other existing degraded conditions were appropriately addressed with respect to their collective impact on continued safe plant operation; and (4) where compensatory measures were involved, whether the measures were in place, would work as intended, and were appropriately controlled. The following evaluations were reviewed:

- CR-GGN-2001-0565, Standby liquid control Pump A discharge bellows failure

- CR-GGN-2001-0505, Standby diesel generator starting air moisture

- CR-GGN-2001-0955, Residual heat removal Train A minimum flow line leakage

- CR-GGN-2001-0596, Standby liquid control Pump A failure

- CR-GGN-2001-1106, Control room fresh air Train A and control room air conditioner Train A

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

The inspectors evaluated significant operator workarounds relative to: (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) whether operator workaround problems were captured in the licensee's corrective action system. The following operator workaround was reviewed:

- Compensatory measures, for a disabled control room annunciator, for the drywell fission product radiation monitor

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

The inspectors reviewed Engineering Request ER-GG-2001-0019, Revision 0, “Raise High Reactor Water Level Trip of the Reactor Feed Pumps and Main Turbine.” The inspectors reviewed the following attributes associated with the modification: (1) that the design bases, licensing bases, and performance capability of the component had not been degraded as a result of the modification; (2) the modification did not place the reactor plant in any unsafe conditions; and (3) adequate postinstallation testing was performed to verify the modification functioned as expected.

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 selected risk significant mitigating systems, to assess whether: (1) the effect of testing on the plant had been adequately addressed by control room and engineering personnel; (2) testing was adequate for the maintenance performed; (3) acceptance criteria was clear and adequately demonstrated operational readiness, consistent with design and licensing basis documents; (4) test instrumentation had current calibrations, ranges, and accuracy for the application; (5) tests were performed, as written, with applicable prerequisites satisfied; and (6) equipment was returned to the status required to perform its safety function. The following activities were reviewed:

- MAI 268930, Division II standby diesel generator operational test following 10-year periodic maintenance
- MAI 288763, Division I standby diesel generator operational test following planned refueling outage maintenance
- MAI 295062, control rod drive hydraulic Pump A operational test following pump replacement
- MAI 285596, RWCU blowdown to main condenser Valve 1G33F235 timing test following valve operator maintenance
- MAI 297233, standby liquid control functional test following standby liquid control Pump A maintenance
- 06-ME-1M61-V-0001, outboard main steam line isolation valves "as left" local leak rate tests following valve maintenance

b. Findings

No findings of significance were identified.

1R20 Refueling and 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:

- Reactor cooldown and transition to shutdown cooling
- Transition from Mode 3 (Hot Shutdown) to Mode 4 (Cold Shutdown)
- Shutdown cooling operations
- Operations with potential to drain the reactor vessel during control rod drive mechanism removal and replacement

- Main steam isolation valve troubleshooting and repair maintenance
- Core alterations
- Reactor vessel hydrostatic test
- Drywell closeout inspection
- Plant safety review committee meeting for startup approval
- Verification and completion of startup prerequisites
- Restoration and heatup of reactor core isolation cooling system
- Alternate decay heat removal verification

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed performance of surveillance test procedures and reviewed test data of selected risk-significant SSCs. The inspectors evaluated whether the SSCs satisfied Technical Specifications, the Updated Final Safety Analysis Report, the technical requirements manual, and licensee procedural requirements. The inspectors also examined whether the testing appropriately demonstrated that the SSCs were operationally ready and capable of performing their intended safety functions. The following tests were inspected:

- 06-OP-1P75-R-0004, Revision 106, “SDG 12, 18 Month Functional Test - Div II LOP/LOCA, and Reject of Single Largest Load”
- 03-OP-1-01-6, Revision 107, “Reactor Vessel In-Service Leak Test”
- 06-IC-1D23-M-0001, Revision 102, “Drywell Atmosphere Radiation Monitor Functional Test”
- STI-GG-2001-0013-00, Revision 105, “Reactor Core Isolation Cooling Valves E51F013 & E51F065 On-Line Local Leak Rate Test”
- 06-OP-1B21-C-0003, Revision 105, “Nuclear Boiler Valve Operability, 18 Month Functional Test, Valve Inservice Testing Requirements”
- 03-1-01-3, Revision 109, “Plant Shutdown Main Steam Line Isolation Valve As Found Stroke Times”

- 06-ME-1M61-V-0001, Revision 107, “Low Flow Air Test Local Leak Rate Test (LLRT)”
- 07-S-74-B21-1, Revision 5, “LLRT Valve Line-up For Main Steam Line Penetration Main Steam Isolation Valves as-found LLRT"
- 06-EL-1L11-R-0003-01, Revision 102, “Division I, 1A3 Battery Bank Service Discharge”
- 06-RE-SB13-V-0401, Revision 104, “In-Sequence Critical Shutdown Margin Demonstration”
- 06-OP-1P75-R-0004, Revision 106, “Division Standby Diesel Generator 24 Hour Load Test / Hot Restart Test”

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed temporary alterations, listed below, to assess the following attributes: (1) the adequacy of the 10 CFR 50.59 evaluation, (2) the fact that the installation was consistent with the modification documentation, (3) the fact that drawings and procedures were updated as applicable, and (4) the adequacy of the postinstallation testing.

- Temporary Alteration 2001-0012, Modification for removal of pulsation dampener bellows in the standby liquid control system Train A pump discharge piping
- Temporary Alteration 2001-0018, Modification for reversing the leads to safety relief Valve 1B21F041D operating solenoid in order to reduce the effect of a ground on the power supply cable

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

On June 20, 2001, the inspector observed a planned licensee emergency preparedness quarterly drill. The inspector reviewed the drill scenario to determine if it reflected realistic plant configurations. The inspector observed licensee personnel at various locations during the exercise, including the control room simulator, technical support center, and emergency operations facility. The inspector 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. The inspector observed the licensee's postdrill critique to determine the adequacy of their drill performance assessment.

b. Findings

No findings of significance were identified.

2 RADIATION SAFETY
Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope

The inspector interviewed radiation workers and radiation protection personnel involved in high dose rate and high exposure jobs during routine operations. The inspector 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:

- Radiation Protection Quality Assurance Audit Report QA-14-2001-GGNS-1 and Quality Assurance Surveillance Reports QS-2000-GGNS-16, -17, and -21

- Area postings 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, surveys, and radiation work permits for the following three significant high dose work areas: removal/replacement of incore detectors (2001-1508), replacement of 14 control rod drives (2001-1516), reactor water cleanup valve work (2001-1616)

- 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 operational radiation protection corrective action documents written since October 2000 (10 of these documents were reviewed in detail: 2000-1761 and 2001-0119, -0244, -0527, -0685, -0674, -0712, -0736, -0756, and -0782)

b. Findings

A noncited violation with very low safety significance (Green) was identified for failure to perform a radiological survey. On April 17, 2001, an NRC inspector noted that the radiation levels indicated on the radiological survey map posted outside of the Residual Heat Removal Pump A room, dated February 21, 2001, ranged from 5-40 millirem per hour. The room was posted as a “High Radiation Area, Contact RP Prior to Entry.” The inspector questioned the posted radiological conditions because the survey information was 2 months old and plant conditions had changed since that time. Prior to entering the room, the inspector contacted radiation protection personnel and was informed that the posted survey map indicated the current radiological conditions in the room.

The inspector entered the room with a radiation survey meter, determined that radiation levels were significantly higher than the levels indicated on the posted map, and left the area. Subsequent survey results of the Residual Heat Removal Pump A room indicated that general area dose rates had increased up to 120 millirem per hour. From discussions with operations personnel, the inspector determined that a plant operator entered the above room earlier in the day and used the map posted outside the room to determine radiological conditions in accordance with station procedures.

The safety significance of this finding was determined to be very low by the Occupational Radiation Safety SDP because there was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised. The issue was more than minor because the failure to perform a radiological survey has a credible impact on safety and has the potential for unplanned or unintended dose.

10 CFR 20.1501(a) states, in part, each licensee shall make or cause to be made surveys that are reasonable under the circumstances to evaluate radiation levels, concentrations, or quantities of radioactive material and the potential radiological hazards. The failure to perform a radiological survey of the above area prior to personnel entering is a violation of 10 CFR 20.1501(a). This violation is being treated as a noncited violation consistent with Section VI. A.1 of the NRC Enforcement Policy. This violation is in the licensee’s corrective action program as Condition Report CR-GGN-2001-0674 (NCV 50-416/2001-03-02).

On April 18, 2001, the licensee identified a second example of a failure to survey violation, see noncited Violation 50-416/2001-03-07 in Section 4OA7 for details.

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

The inspector interviewed radiation workers and radiation protection personnel throughout the controlled access area and conducted independent radiation surveys of selected work areas. No high exposure jobs or work in high radiation areas was performed during the inspection. The following items were reviewed and compared with regulatory requirements to determine whether the licensee had an adequate program to maintain occupational exposures ALARA:

- ALARA program procedures
- Quality Assurance Audit Report QA-14-2001-GGNS-1, “Radiation Protection Program” and Quality Assurance Surveillance Report QS-2001-GGNS-010, “Quality Assurance Observations and Refuel Outage 11 Radiation Protection Activities”
- Five Radiation Protection Department self-assessments (Control Rod Blade Cutup Project, RF11 Refueling Activities, MSIV Work, Scaffolding, and Under Vessel Work)
- 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
- Five radiation work permit (RWP) packages for refueling outage work activities which resulted in the highest personnel collective exposures during Refueling Outage RF11 (RWP 01-1403, “Vessel Disassembly/Reassembly”; RWP 01-1502, “MSIV Work in Drywell”; RWP 01-1507, “Drywell Valve Work”; RWP 01-1516, “Replace 14 CRDs”; and RWP 01-1616, “RWCU Valves Outside Drywell”)
- Use of engineering controls to achieve dose reductions, including six temporary shielding requests (TSR 00-06, -13, -16, -21, -22, and -26)
- Individual exposures of selected work groups (health physics, operations, mechanical, and instruments and charts)
- Hot spot tracking and reduction program
- Radiological work planning
- ALARA Committee meeting minutes of 2001 (1/11, 2/1, 2/2, 2/6, 2/7, 2/8, 2/21, 3/30, 4/11, 4/16, 4/21, 4/27, 4/29, 4/30, and 5/11)
- Declared pregnant worker dose monitoring controls

- Personnel contamination reports (PCR 01-045, -046, and -047) and personnel contamination event (PCE 01-054)
- A summary of radiological worker performance and ALARA related condition reports written since January 1, 2000, was reviewed (13 of these documents were reviewed in detail: CR-GGN-2001-0216, -0241, -0246, -0631, -0718, -0735, -0868, -0879, -0880, -0881, -0884, -0890, and -0894)
- Job site inspections and ALARA controls

b. Findings

A noncited violation with very low safety significance (Green) was identified for failure to perform a radiological survey of a work area on top of the drywell head prior to allowing work in that area. On June 6, 2001, during the review of a personnel contamination event, the inspector noted that the contamination levels on top of the drywell head were significantly higher than the contamination levels indicated around the flange area of the drywell head. The postdecontamination survey of the drywell head flange area performed at 2 p.m. on May 14, 2001, in the reactor cavity, indicated contamination levels from 40,000 to 300,000 disintegrations per minute per 100 square centimeters. However, the survey conducted on top of the drywell head at 5:20 a.m. on May 15, 2001, following the personnel contamination event, documented contamination levels on top of the drywell head ranging from 30 to 700 millirad per hour. Subsequently, the drywell head was posted as a high contamination area.

The inspector determined that the radiological conditions used to brief a worker and prescribe protective clothing and respiratory protection requirements prior to assigning work on top of the drywell head were from a postdecontamination survey of the drywell head flange area. No survey was conducted of the top of the drywell head work area until after the personnel contamination event. This resulted in a personnel contamination and an internal dose assignment of 10 millirem for a worker.

The safety significance of this violation was determined to be very low by the Occupational Radiation Safety SDP because there was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised. The violation was more than minor because the failure to perform a radiological survey resulted in an unintended dose.

10 CFR 20.1501(a), states, in part, each licensee shall make, or cause to be made, surveys that are reasonable under the circumstances to evaluate radiation levels, concentrations, or quantities of radioactive material and the potential radiological hazards. The failure to evaluate the concentrations or quantities of radioactive material and the potential radiological hazards for the top of the drywell head prior to assigning work in that area is a violation of 10 CFR 20.1501(a). This violation is being treated as a noncited violation consistent with Section VI. A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as Condition Report CR-GGN-2001-1088 (NCV 50-416/2001-03-03).

Cornerstone: Public Radiation Safety

2PS2 Radioactive Material Processing and Transportation (71122.02)

a. Inspection Scope

The inspector interviewed licensee personnel, walked down liquid and solid radioactive waste processing systems, and reviewed program documentation to determine if the licensee is meeting the objective of this cornerstone which is to ensure adequate protection of public health and safety from exposure to radioactive material released into the public domain from routine operations. The following items were reviewed and compared with regulatory requirements:

- Radioactive material processing and shipping procedures
- The status of radioactive waste process equipment that was not operational and/or abandoned in place
- Changes made to the radioactive waste processing systems since the last inspection in September 1999
- Waste stream mixing and/or sampling procedures, methodology for waste concentration averaging, and waste classification procedures
- Radiochemical sample analysis results for each of the radioactive waste streams
- The use of scaling factors and calculations used to account for difficult to measure radionuclides
- Changes in waste stream composition due to changing operational parameters and analysis updates
- Shipment packaging, surveying, labeling, marking, placarding, vehicle checks, emergency instructions, disposal manifest, shipping papers provided to the driver, and licensee verification of shipments
- Transport cask certificates of compliance (9157, 9185, 9208, and 9233) and cask loading and closure procedures
- Transferee’s licenses and state/DOT permits
- Conduct of radioactive waste processing and radioactive material shipment preparation activities
- Training program for the conduct of radioactive waste/material processing, packaging and shipping activities

- Seventeen nonexcepted package shipment records (99-0902, -1001, -1011, -1112, -1117, and -1203; 2000-0103, -0202, -0401, -0502, -0503, -0602, and -1202; 2001-0106, -0203, -0301, and -0311)

- Licensee event reports (LERs), special reports, audits (QPA-32.01-99, QA-15-2000-GGNS-1), surveillances (99-01656, QS-2000-GGNS-002 and -004, QS-2001-GGNS-010) and self-assessments related to the radioactive material and transportation programs performed since the last inspection in September 1999

- Fourteen condition reports (CR-GGN-1999-1010, -1011, -1017, -1078, and -1352; -2000-1295, -1575, and -1688; -2001-0119, -0134, -0141, -0171, -0346, and -0465) written against the radioactive material and shipping programs since the previous inspection in September 1999

b. Findings

Waste Classification

A noncited violation with very low safety significance (Green) was identified for failure to properly classify and manifest a radioactive waste shipment. On May 15, 2001, the inspector identified that the licensee excluded measured Pu-241 analysis results from the RWCU-A resin waste stream scaling factors on February 9, 2001. Radioactive Waste Shipment 2001-0203, containing RWCU-A resin, was classified using those scaling factors, manifested, and shipped on February 12, 2001, without determining or documenting the Pu-241 activity. The licensee confirmed that no other shipments during the inspection period were affected.

The safety significance of this violation was determined to be very low by the Public Radiation Safety SDP because radiation limits were not exceeded, and there was no breach of package during transit, certificate of compliance problem, low level burial ground access problem, or failure to make notifications or provide emergency information. The violation was more than minor because there was a credible impact on safety, and it involved an occurrence in the licensee’s radioactive material transportation program.

10 CFR Part 20, Appendix G, III.A.1, requires that any licensee who transfers radioactive waste to a land disposal facility or a licensed waste collector shall prepare all wastes so that the waste is classified according to 10 CFR 61.55. 10 CFR 61.55 (a)(8) states, in part, that the concentration of a radionuclide may be determined by indirect methods, such as use of scaling factors which relate the inferred concentration of one radionuclide to another that is measured, if there is reasonable assurance that the indirect methods can be correlated with actual measurements. 10 CFR Part 20, Appendix G, I.C.10, states, in part, that the shipper of the radioactive waste shall provide the following information on the uniform manifest: the identities and activities of individual radionuclides in each container and the masses of U-233, U-235, and plutonium.

Because the licensee excluded measured Pu-241 activity from its scaling factors, reasonable assurance was not provided that the indirect method of identifying radionuclides in that waste stream was valid. Therefore, the exclusion of Pu-241 in the waste classification and manifest for Radioactive Waste Shipment 2001-0203 was a violation of 10 CFR Part 20, Appendix G. This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as Condition Report 2001-0994 (50-416/0103-04).

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Baseline Performance Indicators

a. Inspection Scope

The inspectors verified the accuracy and completeness of the data used to calculate and report performance indicator data for the last quarter of 2000 and the first quarter of 2001. The inspectors used Nuclear Energy Institute 99-02, Revision 0, "Regulatory Assessment Performance Indicator Guideline," as guidance and interviewed licensee personnel responsible for compiling the information. The following performance indicators were reviewed:

- Unplanned scrams per 7000 critical hours
- Scrams with loss of normal heat removal
- Unplanned power changes per 7000 critical hours

b. Findings

No findings of significance were identified.

.2 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspector reviewed corrective action program records for Technical Specification required locked high radiation areas, very high radiation areas, and unplanned exposure occurrences since October 2000 to confirm that these occurrences were properly recorded as performance indicators. Controlled access area entries with exposures greater than 100 millirems 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 millirems.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspector reviewed radiological effluent release program corrective action records, LERs, and annual effluent release reports documented since October 2000 to determine if any events exceeded the performance indicator thresholds.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

a. Inspection Scope

The inspectors reviewed repeat maintenance preventable functional failures associated with the Division III emergency diesel generator outside building ventilation fan control circuitry and the corrective actions performed to improve system performance. The inspectors reviewed associated condition reports, corrective action items, and engineering evaluations.

b. Findings

A noncited violation, with very low safety significance, was identified for inadequate corrective actions. On June 3, 2001, the Division II EDG outside building ventilation fan failed to automatically start in fast speed as designed while performing periodic surveillance testing. As a result, the licensee declared the EDG inoperable and entered Limiting Condition for Operation (LCO) 01-0423. The licensee entered this condition in their corrective action program as Condition Report (CR) 2001-1063. The licensee determined that the failure of the fan to start in fast speed was caused by a failed relay in the fan control logic circuit and appropriately classified it as a repeat maintenance preventable functional failure in their maintenance rule program.

The inspector noted that this same failure had previously occurred twice on the Division III EDG on April 4 and August 7, 2000. These failures were entered in the licensee's corrective action system under Condition Report 2000-517 and Condition Report 2000-1121, respectively. On August 25, 2000, a corrective action item for Condition Report 2000-1121 was written to replace the subject relays in both the Division I and II EDG building ventilation systems. After further review, the inspector determined that the relays were not replaced as planned but were only functionally tested. The corrective action was closed out, referencing the maintenance activity that performed only a functional test without any technical justification for not replacing the relay.

Failure to replace the subject relay, known to be susceptible to failure, was considered inadequate corrective action and is a violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action." However, because of the very low safety significance of this condition and that the licensee included this condition in their corrective action program, this violation is being treated as a noncited violation (NCV 05000416/2001-003-05) consistent with Section VI.A.1 of the NRC Enforcement Policy. This finding is in the licensee's correction action program as Condition Report 2001-1072. This violation is considered to have very low safety significance because the diesel was able to perform it's safety function with the fan in slow speed and room temperature below 120°F and because the operators would still have the opportunity to manually shift the fan to fast speed prior to the room reaching 120°F.

4OA3 Event Followup (71153)

(Closed) LER 05000416/2001-01

A Technical Specification LCO time was exceeded due to two sequential equipment failures of the Division I EDG. On February 19, 2001, while performing Procedure 06-OP-1P75-003, Revision 105, Attachment I, "Standby Diesel Generator 11, 18 Month Functional 24 Hour Load Test / Hot Restart Test," a turbocharger jacket water cooling leak developed and the diesel generator was declared inoperable while repairs were made to the piping. Technical Specification 3.8.1, Action B.4, required EDG 11 to be restored to operable status within 72 hours, ending at 1 a.m. on February 22, 2001. The leaking jacket water piping elbow was replaced and the cause of failure was determined to be due to low stress high cycle fatigue (Condition Report 2001-0285).

On February 20, 2001, at 12:45 pm, the surveillance test was recommenced and a fuel oil leak developed on Fuel Injection Pump 7 due to a vent bleed screw which became unthreaded and fell out of it's pump casing. The fuel pump bleed screws are not used by the licensee and are received from the vendor in a seated condition with no adjustment or tightening required for use. The licensee determined the apparent cause of the vent screw falling out was insufficient tightening of the screw upon initial assembly by the vendor driven by normal engine vibrations (Condition Report 2001-0303). The inspectors reviewed the licensee's common cause failure analysis for both equipment failures and concluded there was no relationship. The inspectors considered the licensee's corrective actions appropriate.

As a result of the two equipment failures, the licensee was 10.5 hours away from exceeding their LCO time for the Division I diesel generator. On February 21, 2001, at 2:30 p.m., Entergy Operations Inc. formally requested a one time enforcement discretion from the requirements of Grand Gulf Technical Specifications. Grand Gulf Technical Specifications Section 3.8.1, Action B.4, required an inoperable diesel generator to be returned to operable status within 72 hours while in Mode 1 operation or the unit would have to be placed in at least hot shutdown within 12 hours. Entergy Operations, Inc. requested an extension from 72 hours to 108 hours in order to provide sufficient time to restore the inoperable diesel and prevent a unit shutdown. On February 21, at 3:15 p.m., the NRC verbally approved Notice of Enforcement Discretion 01-4-001 per Section VII.C of the Enforcement Policy and considered the enforcement action to have minimal or no safety impact and no adverse radiological impact on public health and safety. The inspectors verified and validated that the licensee had the required compensatory measures in place while the Notice of Enforcement Discretion was in effect. The licensee subsequently satisfactorily performed the surveillance test and declared the diesel generator operable on February 22, 2001, at 4 p.m. This LER is closed.

4OA6 Management Meetings

Exit Meeting Summary

On April 27, 2001, Dan Carter presented his findings relating to access controls to radiation areas to Mr. J. Venable, General Manager, and his staff. The licensee acknowledged the findings presented.

On May 11, 2001, Claude Johnson presented his inspection findings relating to ISI to Mr. B. Eaton, Vice President, Operations, and other members of licensee management. Licensee management acknowledged the inspection findings.

On May 17, 2001, Jim Dodson presented his findings relating to his radioactive material processing and transportation inspection to Mr. J. Roberts, Nuclear Safety Assurance Manager, and his staff. The licensee acknowledged the findings presented.

On June 8, 2001, Blair Nicholas presented his findings relating to ALARA planning and controls to Mr. W. Eaton, Vice President, Operations, and other members of licensee management. The licensee acknowledged the findings presented.

On July 2, 2001, John Whittemore presented his findings relating to heat exchanger performance to Mr. J. Roberts, Nuclear Safety Assurance Manager, and his staff. The licensee acknowledged the findings presented.

On July 5, 2001, the senior resident inspector presented his findings relating to the resident baseline inspection to Mr. J. Venable, General Manager, Plant Operations, and his staff. The licensee acknowledged the findings presented.

In each case, licensee management also informed the inspectors that no safeguards or proprietary material was examined during the inspection period.

4OA7 Licensee Identified Violations

The following findings of very low safety significance were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as noncited violations:

NCV Tracking Number	Requirement Licensee Failed to Meet
50-416/2001-003-06	Technical Specification 5.7.1 requires that an individual permitted to enter a high radiation area be equipped with a radiation monitoring device that continuously integrates dose. On April 21, 2001, the licensee identified that a worker entered a high radiation area with an electronic dosimeter turned off and was, therefore, unable to integrate dose. This event is described in the licensee’s corrective action program, reference Condition Report CR-GGN-2001-0736. The safety significance of this finding was determined to be very low by the Occupational Radiation Safety SDP because there was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised.
50-416/2001-003-07	10 CFR 20.1501(a) states that each licensee shall perform surveys that are reasonable to evaluate radiation levels and potential radiological hazards. On April 18, 2001, the licensee identified that during valve maintenance a worker was in a localized area of a larger room that had not been surveyed. This event is described in the licensee’s corrective action program, reference Condition Report CR-GGN-2001-0672. This violation is being treated as a noncited violation. The safety significance of this finding was determined to be very low by the Occupational Radiation Safety SDP because there was no overexposure, or substantial potential for an overexposure and the ability to assess dose was not compromised.
50-416/2001-003-08	Technical Specification 5.4.1 requires procedures be established, implemented, and maintained covering routine preventive maintenance for safety-related equipment. On April 9, 2001, the licensee's root cause analysis determined that their failure to perform vendor recommended routine inspection maintenance of standby liquid control Pump A contributed to the pump's failure and being inoperable as documented in the licensee's corrective action program in Condition Report 2001-0596. This issue is more than minor because standby liquid control Pump A was inoperable for 5 days. This issue was of very low safety significance (Green) because standby liquid control Pump B was available during that time.

ATTACHMENT

PARTIAL LIST OF PERSONS CONTACTED

C. Bottemiller, Manager, Plant Licensing
B. Edwards, Manager, Maintenance
C. Ellsaesser, Manager, Corrective Action and Assessment
F. Guynn, Manager, Emergency Preparedness
T. Holcombe, Assistant Manager, Operations
C. Lambert, Director, Engineering
R. Moomaw, Manager, Outage Planning and Scheduling
J. Venable, General Manager, Plant Operations
A. Burks, Specialist, Radiation Protection
N. Edney, Supervisor, Radiation Protection
D. Welling, Manager, Technical Support
C. Abbott, Supervisor, Quality Audit
C. Brooks, Senior Licensing Specialist, Plant Licensing
B. Eaton, Vice President, Operations
J. Roberts, Director, Nuclear Safety Assurance
T. Thurmon, Senior Lead Engineer/Maintenance Rule Coordinator, Engineering
H. Yeldell, Manager, System Engineering
D. Barfield, Manager, Design Engineering
K. Christian, Supervisor, Code Program
G. Coker, Quality Assurance Specialist
M. Cross, L-III Nondestructive Examiner
A. Goel, Senior Engineer, Nuclear Safety Assurance
C. Holifield, Senior Engineer, Nuclear Safety Assurance
R. Jackson, Senior Licensing Specialist, Nuclear Safety Assurance
B. Lee, Supervisor, Inspection/Nondestructive Examination
G. Pierce, Director, Program Oversight (Corporate)
M. Renfroe, Manager, Engineering Programs &Components
G. Sparks, Manager, Operations
P. Barnes, Specialist, Licensing
D. Cotton, Supervisor, Radiation Protection
R. Wilson, Superintendent, Radiation Protection
E. Wright, Specialist, Radiation Protection

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened/Closed

05000416/2001-003-01	NCV	Failure to promptly correct a condition that resulted in a degraded heat exchanger surveillance test (Section 1R07)
05000416/2001-003-02	NCV	Failure to perform a radiological survey of the residual heat removal pump A room (Section 2OS1)

05000416/2001-003-03	NCV	Failure to perform a radiological survey on the drywell head (Section 2OS2)
05000416/2001-003-04	NCV	Failure to properly classify and manifest a radioactive waste shipment (Section 2PS2)
05000416/2001-003-05	NCV	Failure to perform corrective action in a timely manner (Section 4OA2)
05000416/2001-003-06	NCV	Failure to wear a proper radiation monitoring device (Section 4OA7)
05000416/2001-003-07	NCV	Failure to perform a radiological survey for valve maintenance (Section 4OA7)
05000416/2001-003-08	NCV	Failure to perform vendor recommended maintenance (Section 4OA7)

Closed

05000416/2001-001	LER	Exceeding Division I EDG time allowed for continued operation with Notice of Enforcement Discretion (Section 4OA3)

LIST OF DOCUMENTS REVIEWED

Procedures:

17-S-06-22, "SSW A Thermal Performance Test," Revision 6
17-S-06-24, "SSW C Thermal Performance Test," Revision 2
06-OP-1P41, "Standby Service Water Loop A Operability Test," Revision 106
01-S-01-3, "Plant Safety Review Committee," Revision 105
01-S-08-, "Exposure and Contamination Control," Revision 108
01-S-08-1, "Administration of GGNS Radiation Protection Program," Revision 102
01-S-08-27, "Radiological Practices for Controlled Areas," Revision 7
01-S-08-34, "Radiological Work Planning, Performance, and Reviews," Revision 1
01-S-08-8, "ALARA Program," Revision 16 and 17
01-S-17-22, "Maintenance Rule Program," Revision 3
01-S-18-6, "Risk Assessment of Maintenance Activities," Revision 0
04-1-01-E12-1, "Residual Heat Removal System," Revision 116
04-1-01-E22-1, "High Pressure Core Spray System," Revision 104
04-1-01-E51-1, "Reactor Core Isolation Cooling System," Revision 115
08-S-01-28, "Use and Control of Temporary Shielding," Revision 10
08-S-02-109, "Coverage and Control of Diving Operations," Revision 4
17-S-06-22, "Standby Service Water A Performance," Revision 6
DT-03-02, "Engineering Support Desktop Maintenance Rule Guide," Revision 19
EDP-045, "Design Engineering Desktop Procedure," Revision 0
LI-102, "Corrective Action Process," Revision 0

01-S-06-2, “Conduct of Operations,” Revision 33
03-1-01-1, “Cold Shutdown to Generator Carrying Minimum Load,” Revision 117
QAI 9.30, “Liquid Penetrant Examination,” Revision 5
QAI 9.15, “Magnetic Particle Examination,” Revision 7
NDE 9.24, “Manual Ultrasonic Examination of Reactor Vessel Ligament Areas,” Revision 2
NDE 9.26, “Ultrasonic Manual Examination of Class 1 Reactor Vessel Welds,” Revision 2
NDE 9.23, “Ultrasonic Examination of Austenitic Piping Welds,” Revision 2
NDE 9.07, “Straight Beam Ultrasonic Examination of Bolts and Studs,” Revision 2
NDE 9.25, “Manual Ultrasonic Examination of Nozzle Radii,” Revision 2
NDE 9.04, “Ultrasonic Examination of Ferritic Piping Welds,” Revision 2
NDE 9.55, “Radiographic Examination of ASME, ANSI, AWS, API, AWWA Welds and Components,” Revision 0
CEP-CII-003, “General Visual Examinations of Class MC Components,” Revision 0
NMMDC-116, “ER Response Installation,” Revision 0
DC-116, “Engineering Request Response Installation,” Revision 0
07-S-07-30, “Maintenance Procedure Welding Documentation Requirements,” Revision 7
17-S-05-15, “Performance and System Engineering Instruction Inservice Inspection,” Revision 3
17-S-14-376, “General Maintenance Instruction Removal and Installation of Limitorque Type SMB/SB and HBC Actuators,” Revision 7

Condition Reports

CR-GGN-1999-1757	CR-GGN-2001-0386	CR-GGN-2001-0049	CR-GGN-2001-0309
CR-GGN-1999-1834	CR-GGN-2001-0682	CR-GGN-2001-0202	CR-GGN-2001-0467
CR-GGN-2001-0120	CR-GGN-2001-0774	CR-GGN-2001-0511	CR-GGN-2001-0828
CR-GGN-2001-0179	CR-GGN-1997-0309	CR-GGN-2001-0548	CR-GGN-2001-0968
CR-GGN-2001-0185	CR-GGN-1998-1195	CR-GGN-2001-0778	CR-GGN-2001-0969
CR-GGN-2001-0293	CR-GGN-2001-0042	CR-GGN-2001-0998	CR-GGN-2001-0961

NDE Documentation

Component	Method of Exam
B33G10-BI-H	UT (50% coverage)
B33G10-BI-F	UT (50% coverage)
B21G012W49	UT
E13N08-1R	UT
E13N07-KA	UT

Miscellaneous:

Calculation MC-01P41-86007, “Standby Service Water Ultimate Heat sink Performance,” Revision 0

Drawing E-1801, “Auxiliary Building and Containment Elevation 139' Fire and Smoke Detection Systems,” Revision 11