UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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ARLINGTON, TEXAS 76011-8064

October 25, 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 50-416/01-04

Dear Mr. Eaton:

On September 29, 2001, the NRC completed an inspection at your Grand Gulf Nuclear Station. The enclosed report documents the inspection findings which were discussed on October 9, 2001, with you 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.

No findings of significance were identified.

Since September 11, 2001, Grand Gulf Nuclear Station has assumed a heightened level of security based on a series of threat advisories issued by the NRC. Although the NRC is not aware of any specific threat against nuclear facilities, the heightened level of security was recommended for all nuclear power plants and is being maintained due to the uncertainty about the possibility of additional terrorist attacks. The steps recommended by the NRC include increased patrols, augmented security forces and capabilities, additional security posts, heightened coordination with local law enforcement and military authorities, and limited access of personnel and vehicles to the site.

The NRC continues to interact with the Intelligence Community and to communicate information to Entergy Operations, Inc. In addition, the NRC has monitored maintenance and other activities which could relate to the site's security posture.

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/

Claude Johnson, Chief
Project Branch A
Division of Reactor Projects

Docket: 50-416
License: NPF-29

Enclosure:
NRC Inspection Report
50-416/01-04

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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No:	50-416
License No:	NPF-29
Report No:	50-416/01-04
Licensee:	Entergy Operations, Inc.
Facility:	Grand Gulf Nuclear Station
Location:	Waterloo Road Port Gibson, Mississippi 39150
Dates:	July 1 through September 29, 2001
Inspectors :	T. Hoeg, Senior Resident Inspector R. Deese, Resident Inspector P. Goldberg, Reactor Inspector R. Mullikin, Senior Reactor Inspector M. Murphy, Senior Reactor Engineer D. Schaefer, Reactor Safeguards Inspector T. Stetka, Senior Reactor Engineer J. Taylor, Reactor Inspector
Approved By:	Claude Johnson, Chief, Project Branch A Division of Reactor Projects

ATTACHMENTS: Supplemental Information

SUMMARY OF FINDINGS

IR 05000416-01-04, on 07/01-09/29/2001; Entergy Operations, Inc., Grand Gulf Nuclear Station. Integrated resident & regional inspection report; Permanent Plant Modifications; Access Authorization Security and Controls; Licensed Operator Requalification Training.

The inspection was conducted by resident inspectors and regional reactor inspectors. The inspection did not identify any violations. The significance of any findings are indicated by their color (Green, White, Yellow, or Red) using IMC 609 "Significance Determination Process." (SDP) Findings for which the Significant Determination Process 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

None

B. Licensee Identified Violations

Two violations of very low safety significance were identified by the licensee and 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 August 7, 2001, when it received an automatic scram resulting from a load reject signal caused by an electrical distribution grid disturbance. The reactor plant was restarted on August 9th and reached 100 percent power on August 11. On August 23, the plant was shutdown for a forced outage to replace a recirculation pump shaft seal assembly. On August 26, the plant was restarted and reached approximately 35 percent power before having to shut down on August 29 to troubleshoot and repair the A recirculation flow control valve control circuitry. On August 30, the plant was restarted and reached 100 percent power where it remained throughout this inspection period.

1. REACTOR SAFETY

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

1R04 Equipment Alignment (71111.04)

.1 Partial System Walkdowns

a. Inspection Scope

The inspectors performed partial system walkdown inspections and reviews 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, system control room indications, and verified valves, breakers, and control circuits were in their required positions for operability. The following systems were inspected:

- Residual heat removal system Train B
- Standby liquid control system Train A
- Division I emergency diesel generator

b. Findings

No findings of significance were identified.

.2 Semi-Annual System Walkdown

a. Inspection Scope

The inspectors performed a complete walkdown of Train B of the standby liquid control system to identify any discrepancies between the existing equipment lineup and the required lineup.

During the walkdown, System Operating Instruction 04-1-01-C41-1, "Standby Liquid Control System," Revision 110, applicable annunciator response procedures, and Drawing M-1082, "P&I Diagram - Standby Liquid Control System," Revision 25, were used to verify major system components were correctly labeled, lubricated, cooled, and ventilated. The inspectors also reviewed open condition reports on the system for any deficiency that could affect the ability of the system to perform its 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 these areas to assess the material condition and operational status of fire detection, suppression systems and equipment; the material condition of fire barriers; and control of transient combustibles. Specific risk significant areas included:

- Auxiliary building electrical penetration room 1A410
- Division III emergency diesel generator room 1D306
- Division II emergency diesel generator room 1D308
- Division I emergency diesel generator room 1D310
- Reactor core isolation cooling system room 1A104
- Division II control building switchgear room OC215

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors reviewed the documentation associated with the thermal performance testing of fuel pool cooling heat exchangers A and B. Specifically, the inspectors reviewed Maintenance Action Items (MAI) 274993, 274994 and supporting documentation. These MAIs were the controlling documents for the fuel pool cooling heat exchanger thermal performance tests performed on April 21, 2001 and April 27, 2001, respectively. In this inspection effort, the test acceptance criteria were reviewed, as well as the licensee's translation of test data to design conditions. The inspectors also evaluated the ability of the licensee's testing frequency to detect degradation prior to loss of heat removal capabilities below design-basis values. It was also noted by the inspectors that the test results considered test instrument inaccuracies and uncertainties and verified the calculations.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Process (71111.11)

a. Inspection Scope

The inspectors: (1) evaluated examination security measures and procedures for compliance with 10 CFR 55.49; (2) evaluated the licensee's sample plan for the written examinations for compliance with 10 CFR 55.59 and NUREG-1021, as referenced in the facility requalification program procedures; and, (3) evaluated maintenance of license conditions for compliance with 10 CFR 55.53 by review of facility records, procedures, and tracking systems for licensed operator training, qualification, and watchstanding. The inspectors also reviewed remedial training and examinations for examination failures for compliance with facility procedures and responsiveness to address areas failed.

In addition, the inspectors: (1) interviewed nine personnel (four operators, two instructors/evaluators, and three managers) regarding the policies and practices for administering examinations; initiating and incorporating feedback from in house and industry events; developing and administering remedial training and retake examinations; (2) observed the administration of two dynamic simulator scenarios to two requalification crews by facility evaluators, including an operations department manager, who led the crew and individual evaluations; (3) observed four facility evaluators administer nine job performance measures, and (4) observed the administration of the biennial written examinations. Each job performance measure was observed being performed by an average of two requalification candidates. The inspectors also reviewed the remediation process for the last training cycle.

The inspectors reviewed the results of the annual and biennial requalification examinations to determine if the results reflected any findings.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

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 the most recent system health reports and system functional failures for the last two years. The following SSCs were reviewed:

- Instrument air system
- Recirculation system
- Fire protection system
- Reactor core isolation cooling system
- Condensate and refueling water storage tank and transfer system
- Low pressure core spray 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 maintenance items reviewed during this period included:

- MAI 299907, Train B Standby Liquid Control Modification
- MAI 302214, Train A Drywell Purge Compressor Corrective
- MAI 299073, Train A Containment Air Cooler Clean and Inspect
- MAI 301601, Division I Emergency Diesel Generator Functional Test

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Plant Events (71111.14)

.1 Automatic Scram

a. Inspection Scope

On August 7, 2001, Grand Gulf Nuclear Station received an automatic reactor scram signal and shutdown. A 500 Kv switchyard disconnect at the Baxter Wilson Station experienced a phase to phase ground causing a grid disturbance sensed by Grand Gulf

Nuclear Station resulting in a generator load reject and subsequent reactor scram. The inspector responded to the plant and observed plant operations personnel placing the reactor plant in a shutdown condition and reviewed nuclear steam supply system responses to the scram. The inspector reviewed Procedures 03-1-01-4, "Scram Recovery," Revision 106, and 03-1-01-3, "Plant Shutdown," Revision 110, and observed licensee actions for procedural compliance.

The inspectors reviewed unexpected system responses including the failure of the end-of-cycle recirculation pump trip (EOC-RPT) instrumentation to initiate as expected, and a hotwell level control system malfunction which caused the condensate pumps to trip on low level and a subsequent loss of normal feedwater control. The inspectors also reviewed licensee actions with regard to 10 CFR 50.72 and 10 CFR 50.73 reporting requirements. Licensee actions were documented in Condition Report 2001-1375.

b. Findings

No findings of significance were identified.

.2 Recirculation Pump B Inboard Seal Failure

a. Inspection Scope

On August 11, 2001, during power ascension and the shifting of recirculation pumps to fast speed, the B recirculation pump inner seal failed with only the outer seal providing the shaft seal function. The inspectors reviewed Operations Standing Order 01-0023 and compensatory measures put in place to monitor, trend, and the required operator actions in response to the failed seal. The inspectors also reviewed the licensee's evaluation for continued operation with the failed seal. The inspectors monitored control room operator personnel for their compliance with Standing Order 01-0023. The licensee documented the seal failure in Condition Report 2001-1386.

b. Findings

No findings of significance were identified.

.3 Forced Outage for Seal Replacement

a. Inspection Scope

On August 23, 2001, Grand Gulf Nuclear Station entered a forced maintenance outage to remove and replace a failed recirculation pump seal and perform other minor maintenance. The inspectors observed portions of the reactor plant shutdown, cooldown, and maintenance planning activities. The inspectors reviewed forced outage radiological work permits associated with the seal replacement including RWP 01-1302, Tasks 1 and 2, for completeness and radiological controls instructions.

b. Findings

No findings of significance were identified.

.4 Exceeding Reactor Core Thermal Limits

a. Inspection Scope

The inspectors reviewed control room operations and events leading up to and including reactor power increases and recirculation pump shifts to fast speed performed on August 27, 2001. The inspectors reviewed the responsibilities of control room reactor engineering personnel and their actions during power increases including recirculation pump shifts and monitoring of core thermal limits. The inspectors reviewed CR 2001-1486 and the licensee's response upon their discovery that the average planar heat generation rate (APLHGR) and linear heat generation rate (LHGR) thermal limit values were above the limits required by Technical Specifications 3.2.1 and 3.2.3, respectively. A licensee identified noncited violation is documented in section 4OA7 of this report.

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-2001-1230, Reactor Core Isolation Cooling (RCIC) System
- CR-GGN-2001-1384, Minimum Critical Power Ratio without Feedwater Heaters
- CR-GGN-2001-1512, End of Cycle Recirculation Pump Trip
- CR-GGN-2001-1476, Main Steam Isolation Valve 1B21FO28A

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

.1 Review of Selected Operator Workarounds

a. Inspection Scope

The inspectors evaluated an operator workaround which required plant operators to manually switch reactor core isolation cooling (RCIC) pump and high pressure core spray (HPCS) pump suctions from the condensate storage tank to the suppression pool in the event of a seismic event. This included examining if the functional capability of the RCIC and HPCS systems were affected or if human reliability in responding to initiating events was affected. The inspectors also specifically evaluated the effect of this operator workaround on the operators' abilities to implement applicable abnormal and emergency operating procedures.

b. Findings

No findings of significance were identified.

.2 Review of the Cumulative Effects of Operator Workarounds

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. The following workarounds were reviewed:

- GGNS Workround No. 1, Radial well flow control valves
- GGNS Workaround No. 3, Seal oil DC pump
- GGNS Workaround No. 4, HPCS/RCIC suction switch over

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17B)

a. Inspection Scope

The inspectors reviewed procedures governing plant modifications to evaluate the effectiveness of the programs for implementing modifications to risk-significant systems, structures, and components, such that these changes did not adversely affect the design and licensing basis of the facility. The inspectors also reviewed 10 permanent plant modification packages and associated documentation, such as 10 CFR 50.59 review screens and safety evaluations, to verify that they were performed in accordance with regulatory requirements and plant procedures. Procedures and permanent plant modifications reviewed are listed in the attachment to this report.

The inspectors interviewed the cognizant design and system engineers for selected modifications as to their understanding of the modification packages.

The inspectors evaluated the effectiveness of the licensee's corrective action process to identify and correct problems concerning the performance of permanent plant modifications. In this effort, the inspectors reviewed corrective action documents (listed in Attachment 1 to this report) and the subsequent corrective actions pertaining to licensee-identified problems and errors in the performance of permanent plant modifications.

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) that equipment was returned to the status required to perform its safety function. The following activities were reviewed:

- MAI 299907, Train B Standby Liquid Control System
- MAI 301103, Control Rod Drive System Flow Recorder
- MAI 298090, A Feedwater Pump Pressure Transmitter
- MAI 293271, Train B Standby Service Water Cooling Tower Flow Transmitter
- MAI 303801, Turbine Building Radiation Monitoring System Alarm

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 to assess whether the SSCs satisfied Technical Specifications, Updated Final Safety Analysis Report, Technical Requirements Manual, 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-ME-1M61-V-0001, "Hydrogen Analyzer Containment Isolation Valve Local Leak Rate Testing," Revision 1

- 06-OP-1P75-M-002, "Division II Emergency Diesel Generator Monthly Functional Test," Revision 2

- 06-OP-1C41-Q-001, "Standby Liquid Control "B" Quarterly Functional Test," Revision 110

- 06-IC-1D17-A-0013, "Turbine Building Radiation Monitor Gaseous Effluent Calibration," Revision 103

- 06-OP-1B21-V-0001, “Main Steam Isolation Valve Functional Stroke Test,” Revision 107

- 06-OP-1P75-M-001, "Division I Emergency Diesel Generator Monthly Functional Test," Revision 2

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) that the installation was consistent with the modification documentation, (3) that drawings and procedures were updated as applicable, and (4) the adequacy of the postinstallation testing.

- No. 2001-018, Automatic Depressurization System FO41D ground
- No. 1998-0021, Lifted leads for protective relay trip circuits

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

On August 20, 2001, 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 licensee personnel at various locations during the exercise including the control room simulator, technical support center (TSC), and emergency operations facility (EOF). The inspectors primarily focused on the ability of the emergency response organization to properly classify the simulated emergency through recognition of emergency action levels (EAL), 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.

3. SAFEGUARDS

Cornerstone: Physical Protection (PP)

3PP1 Access Authorization (71130.01)

a. Inspection Scope

The inspector performed the following inspection activities:

- Reviewed licensee event reports and safeguards event logs to identify problems in the access authorization program
- Reviewed procedures, audits, and self-assessments of the following programs/areas: access authorization and fitness-for-duty
- Interviewed six supervisors/managers and six individuals who had escorted visitors into the protected and/or vital areas to determine their knowledge and understanding of their responsibilities in the behavior observation program
- Reviewed corrective action documents, safeguards event logs, audits, selected security event reports, and self-assessments for the licensee's access authorization program to determine the licensee's ability to identify and resolve problems

b. Findings

No findings of significance were identified.

3PP2 Access Control (71130.02)

a. Inspection Scope

The inspector performed the following inspection activities:

- Reviewed licensee event reports and safeguards event logs to identify problems with access control equipment

- Reviewed procedures and audits for testing and maintenance of access control equipment and for granting and revoking unescorted access to protected and vital areas

- Interviewed security personnel concerning the proper operation of the explosive and metal detectors, X-ray devices, and key card readers

- Observed licensee testing of access control equipment and the ability of security personnel to control personnel, packages, and vehicles entering the protected area

- Reviewed procedures to verify that a program was in place for controlling and accounting for hard keys to vital areas

- Reviewed the licensee's process for granting access to vital equipment and vital areas to authorized personnel having an identified need for that access

- Reviewed corrective action documents, licensee event reports, safeguards event logs, audits, selected security event reports, and self assessments for the licensee's access control program in order to assess the licensee's ability to identify and resolve problems with the access control program

- Interviewed key security department and plant support personnel to determine their knowledge and use of the corrective action reports and resolution of problems regarding repair of security equipment

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES [OA]

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 third and fourth quarters of 2000 and the first and second quarters of 2001. The inspectors used Nuclear Energy Institute 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 0, as guidance and interviewed licensee personnel responsible for compiling the information. The following performance indicators were reviewed:

- Reactor coolant system activity

- Safety system functional failures

- Safety system unavailability, heat removal system (reactor core isolation cooling system)

b. Findings

No findings of significance were identified.

.2 Safeguards Performance Indicators

a. Inspection Scope

The inspector reviewed the program for collection and submittal of performance indicator data. Specifically, a random sampling of security event logs and corrective action reports were reviewed for the following program performance areas:

- Protected area security equipment
- Access authorization program performance
- Fitness-for-duty program performance

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up (71153)

.1 (Closed) LER 05000416/2000-07

On October 26, 2000, the licensee identified deficiencies associated with Kaowool nominal 1-hour fire barrier installations in Fire Zone 1A211 (119' aux building hallway). The deficiency consisted of a Division I safe shutdown cable protruding outside of it's fire wrap while the redundant Division II cable's unistrut support was not fully fire wrapped. This condition would have exposed both redundant division cables to the same fire hazard at the same time. These cable trays contain instrumentation and controls for several Division I and Division II ESF switchgear room coolers.

The inspectors documented their review of LER 2000-007 in IR 2001-02, and considered the associated corrective actions and compensatory measures to be acceptable. The licensee determined the apparent cause of the noncompliance to be an inadequate level of detail provided in the Kaowool installation documentation. In IR 2001-02, the inspectors concluded that failure to provide provisions to assure that appropriate quality standards were controlled to prevent deviations from design documentation was a violation of a license condition to meet the requirements of 10 CFR 50, Appendix R, Section III.G. At that time due to the complexity of this issue, the inspectors considered the violation more than minor and required additional inspection, review, and assistance from the regional senior reactor analyst and opened URI 05000416/2001-002-03.

During this inspection period, following discussions with the licensee, performing system walkdowns, and reviewing the licensee's Engineering Report No. GGNS-94-0051, Revision 1, the inspectors determined that this violation was minor and required no further evaluation. The inspectors reviewed the licensee's fire modeling documentation and concluded that the distance between the two division cables and their height above the floor was much greater than what was required for the amount of thermal energy which could be released during a fire of combustible materials in and around the subject fire zone. As a result, there was no actual or credible impact on safety. The licensee took appropriate corrective actions and will continue performing 1-hour fire watches until the condition can be restored to the required design-basis fire ratings as required by their Technical Requirements Manual. This LER and URI 05000416/2001-002-03 are closed.

.2 <u>(Closed) LER 05000416/2000-02</u>

On December 9, 2000, the licensee identified deficiencies associated with Kaowool nominal 1-hour fire barrier installations in Fire Zone OC215 (Division II ESF switchgear room). The deficiencies consisted of Division I standby service water system safe shutdown cable raceways not having been adequately fire wrapped. This condition would have exposed both redundant Division I and II standby service water system control circuits to the same fire hazard at the same time and potentially rendering the ultimate heat sink inoperable.

The inspectors documented their review of LER 2000-002 in IR 2001-02, and considered the associated corrective actions and compensatory measures taken by Grand Gulf Nuclear Station to be acceptable. The licensee determined the apparent cause of the noncompliance to be an inadequate level of detail provided in the Kaowool installation documentation. In IR 2001-02, the inspectors concluded that failure to provide provisions to assure that appropriate quality standards were controlled to prevent deviations from design documentation was a violation of a license condition to meet the requirements of 10 CFR 50, Appendix R, Section III.G. At that time due to the complexity of this issue, the inspectors considered the violation more than minor and required additional inspection, review, and assistance from the regional senior reactor analyst and opened URI 05000416/2001-002-04.

During the months that followed, a Region IV senior reactor analyst and risk analysts from NRR completed a detailed review of the licensee’s risk assessment of inadequately installed Kaowool in the Division II switchgear room. Additionally, the analysts completed Phase 2 and 3 evaluations in accordance with Manual Chapter 0609, “Significance Determination Process.” The licensee’s evaluation and the NRC’s analysts’ evaluation concluded that the issue was of very low safety significance (Green).

<u>Significance Determination Process Phase 2 Analysis</u>

- The fire ignition frequency for the switchgear space was 8.36E-4/year.

- The fire had the potential to result in a loss of the Division II 4160 safety bus.

The loss of the safety bus resulted in a loss of instrument air and could cause a transient without the power conversion system. The review was completed using the draft Phase 2 plant specific worksheet for a loss of the Division II safety bus.

- Systems potentially affected by the fire involved: control rod drive, instrument air, condensate injection, 2 trains of low pressure coolant injection, 2 trains of residual heat removal, 2 trains of standby service water, 1 train of standby liquid control, 1 train of suppression pool makeup, containment venting, and high pressure core spray.

- No credit was given for automatic suppression and manual suppression capability. Both automatic and manual suppression capability met the design requirements for an area with 1-hour fire wrap. Because the wrap was incorrectly installed, certain fire scenarios could occur which damage cabling before the automatic or manual suppression systems mitigate the fire.

- High degradation was assumed for the inadequate 1-hour fire barrier.

- The condition existed for greater than 30 days.

- There is a likelihood that plant operators would be able to recover Division I standby service water following the fire. The damage from the Division II switchgear fire would prevent automatic actuation of the Division I standby service water system. Operators would be able to manually initiate Division I standby service water in the natural draft mode. The recovery of standby service water would restore the containment heat removal function and room cooling for low pressure safety injection and low pressure core spray. The recovery of standby service water would need to occur within 8 hours.

- A loss of instrument air assumes that low pressure injection is not available if containment heat removal is unavailable. For this fire scenario, low pressure core spray and low pressure coolant injection would remain available, and therefore the low pressure injection function would be maintained. Consequently, the operator action for late depressurization was changed from 1 to 3.

Using Appendix F, “Determining Potential Risk Significance of Fire Protection and Post-Fire Safe Shutdown Inspection Findings,” to Manual Chapter 0609, the fire mitigation factor was determined to be -3.08. The approximate frequency was 1 per 1,000 to 10,000 years. The corresponding estimated likelihood rating for a period of greater than 30 days was D.

Using the draft Phase 2 worksheets for Grand Gulf, the loss of the Division II safety related bus was evaluated. HPCS was considered unavailable due to the fire scenario. Low pressure coolant injection (LPI) was considered available even though standby service water was initially unavailable for room cooling or pump seal cooling. Early inventory control with control rod drive (EICRD) was considered unavailable since one of the two control rod drive pumps was affected by the fire scenario. Containment spray

(CHR) was initially considered unavailable due to a dependency on service water. Because service water was recoverable, there was a likelihood of recovery of containment heat removal.

LAC2: HPCS(0) + CHR(0) + LDEP(3) + Recovery of CHR(1) = 4 (Green)
HPCS(0) + RCIC(1) + LPI(4) = 5 (Green)
HPCS(0) + RCIC(1) + DEP(3) = 4 (Green)

Using Table 1, “Categories for Initiating Events,” of Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” to Manual Chapter 0609, the results were 3 Greens. Therefore, the at-power significance of the inadequately installed fire wrap in the Division II switchgear room is of very low safety significance.

Significance Determination Process Phase 3 Analysis

Because the Phase 2 worksheets were not approved for Grand Gulf, the NRC analysts completed a Phase 3 evaluation.

In general, the NRC analysts determined that the licensee’s risk assessment was of reasonable scope, detail and quality for determining the risk significance of this issue. The licensee determined that the change in core damage frequency (CDF) was approximately 5.0 E-7/year, whereas the NRC analysts determined that the change in CDF was approximately 2.5 E-7. The NRC analysts determined the significance was low because of a relatively low fire ignition frequency, a fire severity factor of 0.1, and the potential for recovery of mitigating systems.

The NRC analysts determined that the licensee used optimistic failure probabilities for recovery of the automatic suppression system and failure of manual suppression. The licensee applied a recovery probability of 0.3 for automatic suppression systems, whereas the NRC analysts did not credit recovery of a potentially failed automatic suppression system. The licensee used a nominal failure probability for manual suppression of 0.06 in conjunction with a severity factor of 0.1. The NRC analysts used a failure probability of 0.5 given the greater dependency on manual suppression for a potentially severe fire. The licensee did not credit recovery of failed safe shutdown systems. The NRC analysts assigned a recovery probability of 0.1 because adequate resources, procedures, and time existed to recover failed equipment.

The following scenarios and factors as they applied to the Kaowool issue were evaluated by the NRC analysts:

	Scenario Description	IF	SF	AS	MS	CCDP	CDF
1a	Fire originates from cabinets that do not directly impact Division I cables but propagation to overhead trays is possible. Auto and manual suppression fails. Hot gas layer (HGL) fails Division I and HPCS cables.	8.01E-4	0.1	0.04	0.5	0.825	1.54E-6

1b	Fire originates from cabinets that do not directly impact Division I cables but propagation to overhead trays is possible. Auto suppression is successful. Both Division I and HPCS cables do not fail because of successful auto suppression and no resulting HGL.	8.01E-4	0.1	1.0	1.0	9.06E-4	8.46E-8
2a	HGL that originates from cabinet which directly impacts Division I cables. Auto suppression is not initially effective for Division I cables. Manual suppression within 6 minutes would be successful but fails. HPCS cables do not fail since auto suppression succeeds within 22 minutes.	3.51E-5	0.12	1.0	1.0	4.6E-2	1.94E-7
2b	HGL that originates from cabinet which directly fails Division I cables. Auto suppression fails and manual suppression (which failed at 6 minutes) also fails within 22 minutes. This results in loss of Division I cables, HPCS and Division II equipment. Probability of manual suppression is based on the difference between 6 and 22 minutes.	3.51E-5	0.1	0.04	0.5	0.825	6.95E-8
2c	Potential HGL that originates from cabinets which directly impact Division I cables. Manual suppression is successful in protecting Division I and HPCS cables.	3.51E-5	0.12	1.0	1.0	9.06E-4	3.82E-9
2d	Potential HGL that originates from cabinets which directly impact Division I cables. Manual suppression is not successful in protecting Division I cables within 6 minutes but does succeed in 22 minutes. HPCS is protected. Auto suppression fails.	3.51E-5	0.12	0.04	0.5	4.6E-2	3.88E-9
3	Fire originates from an ignition source where propagation to overhead trays is not possible but Division I cables are within the damage height from a cabinet fire. No potential for HGL formation. HPCS cables do not fail.	7.03E-5	0.12	1.0	1.0	4.6E-2	3.88E-7
4	Fire originates from an ignition source where propagation to overhead trays is not possible. Division I and HPCS cable do not fail.	4.94E-4	0.12	1.0	1.0	9.06E-4	5.37 E-8

Sub Total	2.34E-6
Recovery	0.1
Total	2.34E-7

Based on the Phase 3 evaluation, the NRC analysts concluded that the inadequate installation of Kaowool in the Division II switchgear room was of very low safety significance (Green). This licensee identified violation is documented in section 4OA7 of this report. This LER and URI 05000416/2001-002-04 are closed.

4OA6 Management Meetings

Exit Meeting Summary

On October 9, 2001, the senior resident inspector presented the inspection results to Mr. W. Eaton, Vice President, and his staff who acknowledged the findings.

The results of the licensed operator requalification inspection were presented on August 31, 2001, to Mr. J. Venable, General Manager, and other members of licensee management at an interim exit meeting. The final exit was held by telephone on September 28, 2001, with Mr. T. McIntyre, Training Supervisor, after receipt and review of the results of the annual and biennial requalification examinations. The licensee management acknowledged the findings presented.

The results of the physical protection inspection were presented on August 23, 2001, to Mr. J. Venable, General Manager, and other members of licensee management who acknowledged the findings.

The results of the permanent plant modification inspection were presented on August 17, 2001, to Mr. J. Venable, General Manager, and other members of licensee management who acknowledged the findings.

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 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.

If you deny these noncited violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region IV; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector at the Grand Gulf Nuclear Station facility.

NCV Tracking Number	Requirement Licensee Failed to Meet
50-416/2001-004-01	Operating License Condition 2.C.41 requires Grand Gulf Nuclear Station to meet the requirements of 10 CFR 50, Appendix R, Section III.G.2. Improperly installed Kaowool in the Division II switchgear room did not provide a required nominal 1-hour fire barrier as reported in LER 2000-002, dated January, 22, 2001. The licensee documented this condition in CR 2000-1481. This finding was treated as a noncited violation. This violation was

more than minor because if left uncorrected, it would become a more significant safe shutdown safety system availability concern due to the potential loss of both the Division I and II standby service water systems simultaneously. The issue was of very low safety significance because of the relatively low fire ignition frequency, a fire severity factor of 0.1, and the potential for recovery of the affected mitigating systems.

50-416/2001-004-02

Technical Specification 5.4.1 requires procedures to be established to provide administrative instructions for equipment controls. On August 27, 2001, the licensee failed to provide adequate prescribed instructions to restore thermal limit monitoring capabilities after deliberately inhibiting the monitoring equipment during a planned transient. The licensee documented this condition in CR 2001-1486. This finding was treated as a noncited violation. The finding had a credible impact on safety because it resulted in the licensee not recognizing operation in a condition exceeding Technical Specification thermal limits in a timely manner. The finding was of very low safety significance because although the finding could have affected the integrity of the fuel cladding, the fuel design limits were not approached and exposure time in this condition was within the limiting condition of operation (LCO).

ATTACHMENT

PARTIAL LIST OF PERSONS CONTACTED

Licensee

C. Abbott, Supervisor, Quality Audit
D. Barfield, Manager, Design Engineering
C. Bottemiller, Manager, Plant Licensing
C. Brooks, Senior Licensing Specialist, Plant Licensing
A. Burks, Specialist, Radiation Protection
W. Eaton, Vice President, Operations
N. Edney, Supervisor, Radiation Protection
B. Edwards, Manager, Maintenance
C. Ellsaesser, Manager, Corrective Action and Assessment
F. Guynn, Manager, Emergency Preparedness
C. Lambert, Director, Engineering
R. Moomaw, Manager, Outage Planning and Scheduling
J. Roberts, Director, Nuclear Safety Assurance
T. Thurmon, Senior Lead Engineer/Maintenance Rule Coordinator, Engineering
J. Venable, General Manager, Plant Operations
D. Welling, Manager, Technical Support
H. Yeldell, Manager, System 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
W. Deck, Security Superintendent
J. Graise, Senior Security Coordinator, Entergy
R. Barnes, Manager, Training and Development
M. Bonds, Quality Assurance Auditor
C. Buford, Licensed Operator Requalification Instructor
E. Cresap, Training
D. Fearn, Simulator Support Supervisor
T. McIntyre, Training Supervisor, Operations

NRC

T. Hoeg, Senior Resident Inspector
R. Deese, Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened/Closed

05000416/2001-004-01	NCV	Failure to meet the requirements of 10 CFR 50, Appendix R, Section III.G.2 (Section 4OA7)
05000416/2001-004-02	NCV	Failure to provide prescribed instructions to restore thermal limit core monitoring capabilities (Section 4OA7)

Closed

05000416/2001-002-03	URI	Division I safe shutdown cable was located outside of the Appendix R fire wrap creating a through path for fire impingement on the Division II cable raceway (Section 4OA3.1)
05000416/2001-002-04	URI	Control building Division II ESF switchgear room (Fire Zone 0C215) fire wrap installation was not in accordance with installation documentation (Section 4OA3.2)
05000416/2000-002	LER	Control building Division II ESF switchgear room (Fire Zone 0C215) fire wrap installation was not in accordance with installation documentation (Section 4OA3.2)
05000416/2000-007	LER	Division I safe shutdown cable was located outside of the Appendix R fire wrap creating a through path for fire impingement on the Division II cable raceway (Section 4OA3.1)

LIST OF DOCUMENTS REVIEWED

Procedures:

01-S-06-3	"Control of Temporary Alterations," Revision 30
03-1-01-1	"Cold Shutdown to Minimum Load," Revision 118
03-1-01-3	"Plant Shutdown," Revision 110
03-1-01-4	"Scram Recovery," Revision 106
LI-102	"Corrective Action Process," Revision 1.
02-S-01-4	"Shift Relief and Turnover," Revision 31.
05-1-02-VI-2	"Hurricanes, Tornadoes and Severe Weather," Revision 104

01-S-06-2	"Conduct of Operations," Revision 111
01-S-06-5	"Reportable Events or Conditions," Revision 105
10-S-03-4	"Control of Combustible Material," Revision 12
01-S-10-1	"Fire Protection Plan," Revision 101
01-S-08-8	"ALARA Program," Revision 18
01-S-17-22	"Maintenance Rule Program," Revision 3
01-S-18-6	"Risk Assessment of Maintenance Activities," Revision 0
PL-163	“Operations Expectations and Standards,” Revision 0
OP-101	“Operations,” Revision 3
TQ-202	“Simulator Configuration Control,” Revision 0
01-S-06-2	“Conduct of Operations,” Revision 111
01-S-04-2	“Licensed Operator Requalification Training,” Revision 11
02-S-01-27	“Operation’s Philosophy,” Revision 4
14-S-02-17	“Administration of Annual Examination,” Revision 1
14-S-02-18	“Job Performance Measure Preparer's Guide,” Revision 1
14-S-02-19	“Job Performance Measure Evaluator's Guide,” Revision 1
14-S-02-20	“Preparing, Conducting, and Review of Simulator Evaluations,” Revision 1
14-S-02-21	“Preparers Guide for Simulator Evaluation Scenarios,” Revision 1
01-S-4-26	“Trainee Performance Review,” Revision 6
01-S-16-1	“Plant Change Implementation,” Revision 104

Modification Packages:

NUMBER	DESCRIPTION
ER-GG-1996-0166-001	Replacement of Pumps D17C004,5,6,7 & 11 per Final Disposition of MNCR 278-975
ER-GG-1996-0444-002, R2	Provide a Modification to Control the Current Oil Leakage from the HPCS Diesel Oil Reservoir
ER-GG-1996-0576-000	Modification to Allow Backseating of Motor Operated Valves Having Problems with Packing Leaks (EER 96/6040)
ER-GG-1999-0466-000, R1	Remove Q2P41F006A-A and Replace with a Blind Spool Piece
ER-GG-1999-0531-000	Install Fuse Status Relays in Division III Switchgear
ER-GG-2000-0052-000	Replace Existing SSW Carbon Steel Pumps with Stainless Pumps (1P41C001B)
ER-GG-2000-0074-000	Modify the 4 Inboard MSIV’s to Ensure Valve Integrity
ER-GG-2000-0074-001	Modify the 4 Outboard MSIV’s to Ensure Valve Integrity

NUMBER	DESCRIPTION
ER-GG-2000-0081-000	Install FlushTaps into the RHR Piping System
ER-GG-2001-0158-000	Standby Liquid Control System Discharge Pressure Rating Increase

Engineering Evaluations:

NUMBER	DESCRIPTION
ER-GG-2001-0177-000	Modification of Valve Q1E12F077A
ER-GG-2000-0890-000	Delete Division III Battery Charger Ground Detection Circuit

Maintenance Action Items (MAIs):

250011	279524	297233
267512	286844	300578
275652	287074	300582
275658	299907	301657

Written Examinations:

01 LOR A5
01 LOR B5

Job Performance Measures:

GG-1-JPM-LOR-E1200-02	Shutdown Suppression Pool Cooling
GG-1-JPM-LOR-B3300-03	Shifting Reactor Recirc Pumps to Fast Speed
GG-1-JPM-LOR-EAL00-02	Emergency Classification: RCS Leakage
GG-1-JPM-LOR-C1100-17	Bypass Rod Position
GG-1-JPM-LOR-C1100-11	Rod Operability Surveillance: One Rod Drift - Stuck
GG-1-JPM-LOR-B3300-04	Shifting Reactor Recirc Pumps to Low Frequency Motor Generator (LFMG)
GG-1-JPM-LOR-P8100-02	Parallel Offsite with D/G [Diesel Generator] 13 - Remote
GG-1-JPM-LOR-P8100-04	Parallel Offsite with D/G [Diesel Generator] 13 - Local

GG-1-JPM-LOR-E1200-04	Shutdown Suppression Pool Cooling
GG-1-JPM-LOR-E2100-04	Perform Quarterly Valve Surveillance in the Control Room
GG-1-JPM-LOR-EOP00-Att 26	RPV [Reactor Pressure Vessel] Injection with Fire Water
GG-1-JPM-LOR-B2100-02	Placing Reference Leg Purge in Service
GG-1-JPM-LOR-EOP00-02	Defeat Main Steam Isolation Valve/Main Steam Line (MSIV/MSL) Drain Level 1 Isolation Interlocks
GG-1-JPM-LOR-EAL00-01	Emergency Classification: Fuel Failure
GG-1-JPM-LOR-N3200-05	Unlock Load Reference Control
GG-1-JPM-LOR-E5100-02	Shutdown Reactor Core Isolation Cooling (RCIC)

Scenarios:

GG-1-SES-LOR-00015	Loss of Vacuum/SDV Block, Revision 2
GG-1-SES-LOR-00018	Failed Open Turbine ControlValve/EHC Fluid Leak/Small Recirc Break with Degraded High Pressure Systems, Revision 2
GG-1-SES-LOR-00025	RCIC Isolation/Feedwater Flow Element Failure/Feedwater Line Break in Drywell, Revision 2

Simulator Discrepancy Reports (DR):

DR 00-0029	Pressure Switch Replacements and Setpoint Changes	01/31/2000
DR 99-0225	Replace Recorder in Control Room	11/08/1999
DR 01-0071	Change Feedwater Control System Level 8 and Time Delay Relay Setpoints	05/02/2001
DR 99-0119	Remove 500 KV Switchyard Indication	06/08/1999
DR 98-0205	Update the 500 KV Mimic Panel	10/05/1998
DR 99-0201	Replace Generator Seal Oil Pumps	09/30/1999

Miscellaneous Documents:

Quality Assurance Audit Report QA-4-2000-GGNS-1, “Design Control,” performed August 7, 2000

Quality Assurance Surveillance Report QS-2000-GGNS-012, “Incorporation of Plant Design Changes into Maintenance Training," performed June 26-29, 2000

Licensee Evaluation PSA-A-2001-001, "Evaluation on the Core Damage Frequency Impact of Degraded Kaowool in the Division II Switch gear Room"

Engineering Report GGNS 94-0051, "Documentation of Fire Modeling for Fire Probabilistic Risk Assessment," pages 3-9, 74-82, and 140-146

Requalification Training, Sample Plan 1999-2001

Engineering Request (ER) to Discrepancy Report (DR) and Training Evaluation Action Request (TEAR) Report

TQ-2000-004-GGNS Training Self-Assessment: SOERs, August 20, 2000

Self-Assessment Visit/Followup of GGNS Responses to INPO Accreditation Visit in Operations December 4, 2000

Quality Assurance Audit of Training, October 4, 2000

D Shift Training Summary, 1999 Cycle 4 - 2001 Cycle 4

D Shift Simulator Scenario Documentation Forms for the period September 7, 1999 through August 9, 2001

Remedial Training Records for the period of May 11, 2000, through August 22, 2001

<u>Condition Reports</u>:

CR 2001-1486	CR 1999-1617	CR 2001-1476
CR 2001-1384	CR 2000-1371	CR 2001-1512
CR 2001-1375	CR 2000-1706	CR 2001-1352
CR 2001-1371	CR 2001-0138	CR 2001-1358
CR 2001-1486	CR 2001-0498	CR 2001-1266
CR 2001-1348	CR 2001-0624	CR 2001-1230
CR 2001-0955	CR 1999-1724	CR 2001-1243
CR 1997-0377	CR 1999-1947	CR 2001-0563
CR 1999-1305	CR 2000-0268	CR 2001-1266
CR 1999-1546	CR 2001-1382	CR 2001-1286
CR 1999-1592	CR 2001-1253	CR 2001-1481

CR 2000-1062
CR 2000-1193
CR 2000-1047
CR 2000-1185

LIST OF ACRONYMS USED

APLHGR	average planar heat generation rate
CDF	core damage frequency
CHR	containment spray
DR	discrepancy report
EAL	emergency action level
EICRD	early inventory control with control rod drive
EOC-RPT	end-of-cycle-recirculation pump trip
EOF	emergency operation facility
ER	engineering request
ESF	engineering safety feature
HPCS	high pressure core spray
LCO	limiting condition of operation
LHGR	linear heat generation rate
LPI	low pressure injection
MAI	maintenance action item
NRC	Nuclear Regulatory Commission
RCIC	reactor core isolation cooling
SSC	system, structure and component
TEAR	training evaluation action request
TSC	technical support center