ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

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License No.:	NPF-29
Report No.:	50-416/99-17
Licensee:	Entergy Operations, Inc.
Facility:	Grand Gulf Nuclear Station
Location:	Waterloo Road Port Gibson, Mississippi 39150
Dates:	October 17 through November 27, 1999
Inspectors:	Jennifer Dixon-Herrity, Senior Resident Inspector Peter Alter, Resident Inspector Michael Hay, Resident Inspector, Cooper Nuclear Station
Approved By:	Joseph I. Tapia, Chief, Project Branch A

ATTACHMENT: Supplemental Information

EXECUTIVE SUMMARY

Grand Gulf Nuclear Station
NRC Inspection Report No. 50-416/99-17

This inspection included aspects of licensee operations, maintenance, engineering, and plant support. The report covers a 6-week period of resident inspection.

Operations

- Operations personnel exhibited good oversight and direction of activities during the reduction in power and scram to start Refueling Outage 10 (Section O1.1).

- The operability determination performed to justify continued operability of the alternate decay heat removal subsystem after a transient in the plant service water system was not thoroughly documented. The operators on shift adequately explained why the alternate decay heat removal subsystem was not affected. Although the operability determination was satisfactory, the licensee acknowledged that it was not thoroughly documented (Section O1.2).

- The safety-related areas of the plant were maintained in good condition during the refueling outage (Section O2.1).

Maintenance

- The 12 maintenance activities observed during this period were well conducted, with one exception. An electrical maintenance technician inadvertently caused an isolation of the plant service water system to the auxiliary building while troubleshooting a computer alarm point. The licensee determined that poor understanding of the scope of the troubleshooting work package led to the trip of the auxiliary building isolation valve power supply. Operators were able to restore plant service water to the auxiliary building and the standby decay heat removal system within 2 minutes (Section M1.1).

- Operators successfully demonstrated the alternate decay heat removal subsystem capability within 24 hours of reactor shutdown. However, the sequence of steps in the procedure used to perform the demonstration had the potential to cause an unintended mode change if the steps were not timed properly. The senior reactor operator in charge of the evolution recognized this deficiency and resequenced the procedure steps before the prejob brief. The licensee subsequently revised the procedure (Section M3.1).

Engineering

- The engineering evaluation of the results of in-vessel visual inspections adequately addressed the discrepancies found (Section E1.1).

- The corrective actions taken in response to a damaged bearing shell found in the Division I standby diesel generator were thorough. The engineering evaluation adequately addressed the failure and the potential for common mode failure (Section E1.2).

Plant Support

- Observed activities involving radiological controls were generally well performed. The inspectors identified one example where a high radiation area boundary rope had fallen. The area had been conservatively posted as a high radiation area and the corrective actions for a similar problem identified in Condition Report CR-GGN-1999-1375 should prevent repeat of this problem. The licensee immediately replaced the rope and initiated CR-GGN-1999-1736 to document the incident (Section R1.1).

- The daily security activities during the outage were generally well conducted. The inspectors identified one example where an unattended truck was immobilized with a loosely chained steering wheel. The licensee promptly tightened the steering wheel. The licensee met procedure requirements in that security officers maintained control of the vehicle by chaining the steering wheel and maintaining control of the key. The licensee acknowledged that the method of locking the truck did not meet management expectations and completed a security deficiency report (Section S1.1).

Report Details

Summary of Plant Status

The plant was at 100 percent power at the beginning of the inspection period. On October 21, 1999, the licensee commenced lowering power to approximately 50 percent. On October 22, 1999, the licensee lowered power to approximately 25 percent and operators manually inserted a scram to start Refueling Outage 10. The plant was in Mode 5 at the end of the inspection period.

I. Operations

O1 Conduct of Operations

O1.1 Reactor Shutdown

a. Inspection Scope (71707)

The inspector observed as control room operators decreased reactor power and manually inserted a scram to start Refueling Outage 10.

b. Observations and Findings

On October 21-22, 1999, the inspectors observed as operators decreased power in a controlled manner using control rods. The operators inserting the rods maintained very good control over the evolution, performing self-checks and independent verifications to ensure the correct rods were inserted. Reactor engineering and operations supervisory personnel closely monitored the evolution. Throughout the observed activity, noise level in the control room was well controlled and supervisors limited access to the control room to personnel who had reason to be in the control room.

The briefing held prior to inserting the scram was thorough. At midnight, operators manually inserted a scram. The evolution was planned and each operator had a task to perform. All systems responded as expected and operators maintained good control of the plant.

c. Conclusions

Operations personnel exhibited good oversight and direction of activities during the reduction in power and manual insertion of a scram to start Refueling Outage 10.

O1.2 Operability Determination Documentation Quality

a. Inspection Scope (71707)

The inspector reviewed the operability determination conducted by operations personnel in response to the loss of plant service water to the auxiliary building on October 29, 1999.

b. Observations and Findings

Operations personnel initiated Condition Report CR-GGN-1999-1429 to document the isolation of plant service water to the auxiliary building as a result of a breaker being inadvertently opened during troubleshooting. The maintenance aspects of this event are discussed further in Section M1.1. The isolation of plant service water to the auxiliary building resulted in the mechanical failure of several pressure retaining components in radial well Pumps A, B, J, and K. At the time, the plant was relying on the alternate decay heat removal (ADHR) subsystem as the available alternate shutdown cooling source. Plant service water was the cooling source for the ADHR heat exchangers. The operability determination conducted by operations personnel documented that the ADHR subsystem was operable because the number of available plant service water pumps was equal to the number of pumps that were running when the ADHR subsystem was tested to verify that it would effectively maintain the reactor coolant temperature.

The inspectors questioned how the number of pumps had a bearing on the system's operability. The plant service water pumps have different capacities because of the quality of the water and the location of the wells with respect to the river. The total flow available would have to be addressed to verify whether the system would function effectively. In addition, equipment was damaged during the transient. The operability determination did not address the condition of the portion of the system that was still on-line. The inspectors discussed the concerns with the senior reactor operators (SROs) in the control room. Operators had conducted walkdowns of the wells and verified that the pumps that were available were not adversely affected by the transient. In addition, the flow from three of the available pumps was enough to allow the ADHR subsystem to cool at the required capacity.

The inspectors discussed the adequacy of the documented operability determination with the operations manager. The manager acknowledged the concern.

c. Conclusions

The operability determination performed to justify continued operability of the alternate decay heat removal subsystem after a transient in the plant service water system was not thoroughly documented. The operators on shift adequately explained why the alternate decay heat removal subsystem was not affected. Although the operability determination was satisfactory, the licensee acknowledged that it was not thoroughly documented.

O2 Operational Status of Facilities and Equipment

O2.1 Plant Tours

a. Inspection Scope (71707)

The inspectors conducted tours through safety-related portions of the plant.

b. Observations and Findings

The areas of the plant that were toured were maintained in good condition. Early in the inspection period, the licensee was preparing for Refueling Outage 10. The scaffolding and other materials staged for the outage were located so as not to interfere with safe operation. Very little construction was started in safety-related spaces until the outage commenced. Personnel contracted to work during the outage started arriving onsite at the beginning of the period. Although there was a small increase in licensee identified human errors, the transition was handled well.

During the outage, the areas containing equipment being used for shutdown cooling, emergency core cooling, and emergency power were posted with signs indicating that the areas were off limits due to high risk. The inspectors toured these spaces regularly to ensure that the equipment was operable and maintained in good condition and that the signs were being followed. The inspectors observed that a scaffold had been left in the ADHR pump room under the residual heat removal Train C suction valve. The scaffold coordinator found that the work had been completed on the valve before the outage, but indicated that he had received no request to remove it. Although the scaffold was seismically constructed, the inspector observed that there was no requirement for the scaffold and that it should have been removed from the area when the work was completed.

The inspectors toured the drywell on November 7, 1999. For the most part, the contaminated area around the suppression pool and the drywell were maintained in good condition for the amount of work that was occurring. There was a shoe cover, scrap tape, and other trash left on the catwalk where it could be knocked into the suppression pool. The drywell was cluttered and contained materials that had to be removed, but was in better condition than at the same time during the last outage. The licensee had the loose material removed. The licensee was maintaining close control of the condition of the drywell and regularly provided status on area conditions during the morning status meetings.

During a tour of containment on November 17, 1999, inspectors observed water overflowing the containment floor drain sump. The licensee quickly responded by roping off the area as a contamination area. A temporary pump was then installed to drain the tank into the suppression pool. In the control room, the containment floor drain sump Hi/Hi level alarm did not alarm for approximately one hour after the tank overflowed. In addition, the alarm did not appear to clear at the desired setpoint while draining the tank down using a temporary sump pump. Operators in the control room were sensitive to the condition and entered the discrepancy as Maintenance Action Item (MAI) 267715.

The inspector noted that a portion of wire heating strip used on the standby liquid control system piping had become frayed. The inspectors informed the control room of this and operators had electricians determine whether the heating strip was functional. The heating strip was determined to be operable and repairs were made to insulate the exposed wiring.

c. Conclusions

The safety-related areas of the plant were maintained in good condition during the refueling outage.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Maintenance and Surveillance Observations

a. Inspection Scope (61726, 62707)

The inspectors observed all or portions of the maintenance, surveillance, and test activities listed below. Maintenance work was reviewed to ensure that adequate work instructions were provided, that the work performed was within the scope of authorized work, and that the work performed was adequately documented. For surveillances, the test procedures were reviewed and compared to the Technical Specification surveillance requirements and bases to ensure that the procedures satisfied the requirements. In all cases, the impact to equipment operability and applicability of Technical Specification actions were independently verified. The following are the MAIs and surveillance tasks observed:

Maintenance:

- 266504 Troubleshoot Load Center LCC 15BA6
- 250188 Standby service water basin maintenance
- 259478 Bus 15 Maintenance
- 266727 Main steam isolation valve troubleshooting
- 266008 Division I diesel 25 percent tear down
- 266592 Troubleshoot Division I breaker fault computer point
- 261075 Install new relay clips

Surveillance/Test:

- 04-1-03-N21-6 Reactor Feed Pump Turbine A Overspeed Trip Test
- 04-1-01-E12-1 Residual Heat Removal System - Demonstration of ADHR Subsystem
- 06-OP-SZ51-R-003 Control Room Fresh Air Filter Test
- 06-OP-1P75-R-004 Standby Diesel Generator 12: 18 Month Functional Test
- 06-OP-1C41-R-002 Standby Liquid Control Injection Test

b. Observations and Findings

The inspectors observed that the work performed during these activities was well conducted, with one exception. While troubleshooting a computer alarm point for Division I motor control center breaker faults on October 29, 1999, an electrician tripped the breaker for the power supply for the solenoid switches on the Division I auxiliary building plant service water isolation valves. The valves closed, resulting in a loss of plant service water to the ADHR subsystem. At the time, the ADHR subsystem was the alternate method of decay heat removal. The electrician immediately informed the control room and was directed to reclose the breaker. The operators restored plant service water to the auxiliary building and the ADHR subsystem within 2 minutes.

The inspectors reviewed MAI 266592 and interviewed the electrician and on-shift operators. The electrician stated that he did not fully understand the scope of the MAI and was in the process of correcting the problem when he inadvertently tripped the breaker, causing the isolation. The SRO, who authorized the start of work, stated that the troubleshooting was understood to be nonintrusive and, as a result, was not recognized as a high risk evolution for the current shutdown conditions. The licensee wrote Condition Report CR-GGN-1999-1428 to review the event and posted the isolation valve power supply as a high [shutdown] risk impact area.

c. Conclusions

The 12 maintenance activities observed during this period were well conducted, with one exception. An electrical maintenance technician inadvertently caused an isolation of the plant service water system to the auxiliary building while troubleshooting a computer alarm point. The licensee determined that poor understanding of the scope of the troubleshooting work package led to the trip of the auxiliary building isolation valve power supply. Operators were able to restore plant service water to the auxiliary building and the standby decay heat removal system within 2 minutes.

M3 Maintenance Procedures and Documentation

M3.1 ADHR Subsystem Demonstration

a. Inspection Scope (71707)

The inspectors observed the demonstration of the ADHR subsystem as the operating decay heat removal subsystem. The inspectors reviewed Section 5.13, "ADHR - RPV Cooling Mode Startup," of Procedure 04-1-01-E12-1, "Residual Heat Removal System," Revision 112, and the applicable portions of engineering Calculation MC-Q1E12-90010, "Reactor Decay Heat Removal Assessment for ADHR[S]," Revision 0.

b. Observations and Findings

On October 23,1999, operators performed the required ADHR subsystem demonstration within 24 hours of reactor shutdown. The SRO in charge of the evolution determined that the procedure section for placing the ADHR subsystem in service was inappropriate for current plant conditions. The sequence of steps, as written, prolonged the time between removal of the operating residual heat removal system from service and starting of ADHR subsystem cooling. This could have led to an unintended mode change if reactor pressure vessel temperatures exceeded 200°F. The SRO marked up copies of the procedure for the operators involved and conducted a comprehensive prejob brief for the entire operating crew. Each operator was given an opportunity to review the procedure and offer suggestions during the prejob brief. The evolution was successfully completed and demonstrated the results of the engineering calculation for the ADHR subsystem decay heat removal capability. The marked up procedure section was forwarded to the operations section procedure coordinator to allow revision of Procedure 04-1-01-E12-1. The inspectors determined that the action taken by the SRO to re-sequence the procedure steps was appropriate for the circumstances and in accordance with Procedure 02-S-01-2, "Control and Use of Operations Section Directives," Revision 33.

c. Conclusions

Operators successfully demonstrated the alternate decay heat removal subsystem capability within 24 hours of reactor shutdown. However, the sequence of steps in the procedure used to perform the demonstration had the potential to cause an unintended mode change if the steps were not timed properly. The SRO in charge of the evolution recognized this deficiency and resequenced the procedure steps before the prejob brief. The licensee subsequently revised the procedure.

III. Engineering

E1 Conduct of Engineering

E1.1 Evaluation of In-Vessel Visual Inspections

a. Inspection Scope (37551)

The inspectors reviewed the engineering evaluation of in-vessel visual inspections performed during the refueling outage and the operability evaluation of the core spray sparger bracket bolt tack weld cracks documented in Condition Report CR-GGN-1999-1531.

b. Observations and Findings

The inspectors determined that the engineering operability recommendation for the core spray sparger bracket bolt tack weld cracks discovered during in-vessel visual inspections done during the refueling outage was comprehensive and within the scope of a previous evaluation performed for Condition Report CR-GGN-1998-0357. Intergranular stress corrosion cracking and loose parts concerns were appropriately considered. The inspectors determined that the licensee's plans, already in place, to inspect and evaluate the core spray bracket bolt tack welds during each refueling outage were adequate to address the operability of the low pressure core spray system and to meet the requirements of the in-vessel visual inspection program.

c. Conclusions

The engineering evaluation of the results of in-vessel visual inspections adequately addressed the discrepancies found.

E1.2 Disposition of Damaged Bearing Shell

a. Inspection Scope (62707)

The inspectors reviewed the licensee's disposition of a damaged Division I standby diesel generator lower bearing shell.

b. Observations and Findings

Upon removing the rods from Cylinders 4, 6, and 8 as part of the scheduled 10-year inspection of the Division I standby diesel generator, the licensee found that the lower bearing shell on Rod 4 was damaged and that there was a small crack in the upper bearing shell. The lower shell had a section of material approximately 2 inches by 4 inches that was loose in the center of the shell. The crack in the upper shell was approximately one inch long and was located on an oil groove centerline. The piece was restrained such that it could not have fallen out. The licensee had the rods removed from Cylinders 3 and 5 to determine if further damage was present. No further damage was observed on any other rod bearing shells.

The licensee forwarded the damaged shells to a welding test laboratory to have the material metallurgically tested and had the manufacturer perform a root cause analysis. The manufacturer determined that the most probable cause was an event that occurred in May 1995. At that time, the left bank air inlet butterfly valve failed closed during a 100 percent load run. The right bank cylinders picked up most of the load. During this time, the Rod 4 bearing experienced some degree of overheating, causing the bearing stress load to exceed the bearing stress limit and potentially initiating the crack. An internal inspection of the engine at the time did not reveal any damage to the engine. The manufacturer determined that the operability of the engine was not affected. Industry experience in commercial applications was that cracked bearing shells caused no ill effects on engine operation. Due to the condition of the other four sets of rod bearing shells inspected and the licensee's review of the vendor's evaluation, the licensee determined that the bearing shell failure was an isolated event and required no further investigation.

c. Conclusions

The corrective actions taken in response to a damaged bearing shell found in the Division I standby diesel generator were thorough. The engineering evaluation adequately addressed the failure and the potential for common mode failure.

IV. Plant Support

R1 Radiological Protection and Chemistry Controls

R1.1 Radiation Protection During the Refueling Outage

a. Inspection Scope (71750)

During tours of the radiologically controlled area, the inspectors observed radiological postings and worker adherence to radiation protection procedures.

b. Observations and Findings

The inspectors observed that personnel followed radiation protection procedures, locked high radiation area doors were locked, and, in all but one instance, radiation and contamination areas were properly posted and barricaded. On November 18, 1999, the inspectors identified a high radiation area that was not properly barricaded. The high radiation area surrounded a portion of residual heat removal system piping on the 135-foot level of containment. The inspector noted that the area was properly posted; however, one of the ropes used to barricade the area was down. The licensee performed a radiation survey of the area and determined that radiation levels in the area were approximately 38 mrem per hour on contact and 15 mrem per hour one foot from the source. The inspector was informed that the area was posted as a high radiation area due to the potential for radiation levels in that area to change as plant conditions changed.

Technical Specification 5.7.1 stated, in part, each high radiation area, as defined in 10 CFR Part 20, in which the intensity of radiation is greater than 100 mrems per hour but less than 1000 mrems per hour, shall be barricaded. The inspectors determined that the failure to maintain the barricade around the residual heat removal piping was similar to noncited Violation 50-416/9913-01. This noncited violation was documented in Condition Report CR-GGN-1999-1375 and the corrective actions had not been completed. The licensee planned to take further corrective actions to specifically address this example in conjunction with corrective actions for the previously identified noncited violation. On November 18, 1999, the licensee documented the problem in CR-GGN-1999-1736.

c. Conclusions

Observed activities involving radiological controls were generally well performed. The inspectors identified one example where a high radiation area boundary rope had fallen. The area had been conservatively posted as a high radiation area and the corrective actions for a similar problem identified in Condition Report CR-GGN-1999-1375 should prevent repeat of this problem. The licensee immediately replaced the rope and initiated CR-GGN-1999-1736 to document the incident.

S1 Conduct of Security and Safeguards Activities

S1.1 Security During the Outage

a. Inspection Scope (71750)

On a daily basis, the inspectors observed security personnel practices and the condition of security equipment during the outage.

b. Observations and Findings

The inspectors observed that security maintained good controls at the access point and that areas affected by the outage were posted with a security guard as required. On October 28, 1999, the inspectors observed that a truck being used to drain the standby service water basin had been left unattended while running. The steering wheel was loosely chained to a long handle at the door of the truck. The inspectors discussed the concern with a security supervisor to determine if this was the normal practice for securing vehicles. The supervisor found that the steering wheel could be turned completely around and stated that it should have been immobilized.

The inspector noted that the wheels of the truck turned a little less than 45 degrees with one turn of the steering wheel. This would have allowed the truck to be steered into the fence or into the maintenance building. The officers explained that the normal practice for immobilizing a vehicle was to turn the wheel as far in one direction as possible, then to lock the wheel tightly. In this case, the truck could not be locked by turning the wheel as far as it would go because it would have damaged the power steering system, but it could have been locked with a shorter chain and to a smaller connection. The supervisor explained that it did not meet his expectations and he had the truck's steering wheel locked more securely and had the officers verify that the second truck being used at the basin was properly secured. The inspector found that Procedure 11-S-11-4, "Vehicle Control," Revision 2, required that the security force use means to control the vehicle. One example provided was to chain and padlock the steering wheel where the key to the lock was controlled by the security force. The licensee documented the concern in a security deficiency report.

c. Conclusions

The daily security activities during the outage were generally well conducted. The inspectors identified one example where an unattended truck was immobilized with a loosely chained steering wheel. The licensee promptly tightened the truck steering wheel. The licensee met procedure requirements in that security officers maintained control of the vehicle by chaining the steering wheel and maintaining control of the key. The licensee acknowledged that the method of locking the truck did not meet management expectations and completed a security deficiency report.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on December 2, 1999. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT

PARTIAL LIST OF PERSONS CONTACTED

Licensee

C. Bottemiller, Manager, Plant Licensing
B. Edwards, Manager, Planning and Scheduling
C. Ellsaesser, Manager, Corrective Action and Assessment
C. Stafford, Manager, Plant Operations
J. Venable, General Manager, Plant Operations
R. Wilson, Superintendent, Radiation Protection

INSPECTION PROCEDURES USED

37551	Onsite Engineering
61726	Surveillance Observations
62707	Maintenance Observation
71707	Plant Operations
71750	Plant Support Activities

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Closed

None