ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Inspection Report: 50-416/96-02

License: NPF-29

Licensee: Entergy Operations, Inc.
P.O. Box 756
Port Gibson, Mississippi

Facility Name: Grand Gulf Nuclear Station

Inspection At: Port Gibson, Mississippi

Inspection Conducted: January 8 through February 9, 1996

Inspectors: Thomas F. Stetka, Senior Reactor Inspector, Engineering Branch, Division of Reactor Safety

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Approved: Chris A. VanDenburgh, Chief, Engineering Branch, Division of Reactor Safety — 4-5-96 Date

Inspection Summary

Areas Inspected: Routine, announced inspection of the licensee self assessment of the engineering and corrective action programs.

Results:

Engineering

- The NRC inspectors concluded that the licensee's self assessment encompassed all the inspection requirements of NRC Inspection Procedures 37550 and 40500; therefore, no additional core inspection was required (Section 2.2).

- The self-assessment team concluded that no safety degradations or departures from regulatory requirements were identified during the course of the assessment. Findings that were identified as quality deficiencies were documented in the appropriate corrective action document by the licensee. In contrast to the self-assessment team's conclusion, the NRC inspectors concluded that minor departures from regulatory requirements occurred (Section 3.1).

- The licensee's revision to the surveillance test procedure for venting the residual heat removal system was not timely (Section 3.3.1).
- System engineers felt that they had little time for non-immediate issues because of their involvement with daily work demands (Section 3.3.2).
- Despite similar failures at other Entergy Operations facilities, the licensee had no plans to inspect or replace the reactor core isolation cooling pump turbine governor valve stem, which was susceptible to corrosion, during the next outage (Section 3.3.3).
- Operation of the plant service water system in the manual mode was considered by the NRC inspectors to be an operator work around (Section 3.3.5).
- Plant material conditions were very good in heavily trafficked areas and only adequate in areas not easily accessed or trafficked (Section 3.4).
- System reliability for the reactor core isolation cooling system and the residual heat removal system considerably exceeded the licensee's goals (Section 3.4).
- Operations personnel were very satisfied with the support that they received from the engineering organizations (Section 3.4).
- The self-assessment team concluded that some discrepancies were noted between licensing, training, and plant documents (Section 3.5).
- The design organization was effective at supporting the plant in assuring that the automatic depressurization system was capable of performing its safety function (Section 3.6).
- System engineers were found to be knowledgeable of their systems and of the open items related to their systems (Section 3.6).
- Licensee personnel effectively managed engineering work backlogs and prioritization of work activities (Section 3.7.4).
- The licensee's self-assessment team was very effective in conducting the assessment (Section 3.8.1).

- While the corrective action process for identifying, resolving, and preventing problems provided timely identification, cause determination, and corrective actions, the process did not appear to be effective in identifying problems that were precursors to new problems (Section 3.8.2).

- The top ten quality issues list was effective in assuring that issues were resolved in a timely manner (Section 3.8.3).

- While the root-cause analysis program was effective, there were instances where root-cause analyses were either too narrow or too broad. In addition, there was evidence of a growing backlog (Section 3.8.4).

- The operating experience feedback program was considered to be effective. It provided comprehensive monthly reports, was disseminating information to site organizations, and was providing timely information to the industry (Section 3.8.5).

Summary of Inspection Findings:

- Three noncited violations were identified (Sections 3.2 and 3.5).

Attachment:

- Attachment - Persons Contacted and Exit Meeting

DETAILS

1 INTRODUCTION (40501)

The purpose of this inspection was to determine the effectiveness of the licensee's self assessment of their engineering and corrective action programs. In a letter dated November 17, 1995, the licensee proposed to perform a self-assessment of the engineering and corrective action programs in accordance with the guidance of NRC Inspection Procedure 40501, "Licensee Self assessments Related to Team Inspections." As is the customary practice in Region IV, a team inspection was planned to accomplish the core inspection program requirements of NRC Inspection Procedures 37550, "Engineering," and 40500, "Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems." The option of permitting licensees to conduct a self assessment in lieu of an NRC team inspection is an NRC program aimed at minimizing regulatory impact and utilizing NRC resources more efficiently. Personnel from the Entergy Operations corporate self-assessment group organized the effort, using their corporate process.

Two NRC inspectors reviewed the self-assessment team's effort from January 8 through February 9, 1996, in accordance with NRC Inspection Procedure 40501. The NRC inspectors observed the performance of the self-assessment team during the first week of onsite inspection. The NRC inspectors performed a second week of onsite independent inspection to ensure the satisfactory completion of the team's self assessment.

2 SELF-ASSESSMENT SCOPE EVALUATION

2.1 Initial Evaluation

In the November 17, 1995 letter, the licensee provided the NRC an outline of the objectives, scope, general approach, schedule, level of effort, and team member's qualifications to be used in the self-assessment. The NRC inspectors compared this outline against the requirements of NRC inspection procedures 37550 and 40500 and determined that the licensee's self-assessment plans included all of the key elements listed in both NRC inspection procedures. The NRC inspectors also reviewed the qualifications of the self-assessment team members. In a letter dated December 8, 1995, the NRC accepted the proposed timing, scope of effort, and the credentials and experience of the licensee's self assessment.

2.2 Implementation

During performance of the self assessment, the NRC inspectors monitored the self-assessment team's activities to determine whether the requirements of NRC Inspection Procedures 37550 and 40500 were being fully addressed. The inspection was conducted over two separate weeks. During the first week, January 8-12, 1996, the NRC inspectors observed the conduct of the in-process self assessment. The observations included system walkdowns, personnel interviews, and record reviews. During the second week, February 5-9, 1996, the NRC inspectors conducted a technical inspection of the completed self assessment using the self-assessment team's completed report. This inspection included verification that all self-assessment findings were included in the licensee's corrective action program and an independent inspection of selected areas. Based upon these observations and reviews, the NRC inspectors determined that the licensee's self assessment fully implemented these NRC inspection procedures.

2.3 System Selection

The NRC inspectors interviewed the self-assessment team leader and reviewed the team's self-assessment report to determine the team's basis for selecting the systems reviewed during the self assessment. The team had selected the automatic depressurization system for the vertical review approach. The team intended that this vertical review would provide an integrated approach to assessing the overall results of engineering activities as they related to maintaining the design basis of the in-plant components. The team selected this system based on risk significance and the fact that this system had not been previously evaluated in any other self assessment or NRC inspection.

The self-assessment team selected four additional systems for a broad review approach. These systems included the residual heat removal, reactor core isolation cooling, plant service water, and standby gas treatment systems. The team selected the residual heat removal system because of its high risk significance and its relationship with the automatic depressurization system as a low pressure injection system. The team selected the reactor core isolation cooling system based on its importance to core damage, its importance in station blackout scenarios, and the industry/NRC initiatives regarding Terry turbines. The team selected the plant service water system (a nonsafety-related system) based on its importance to core damage and recent plant events. Although the last system selected did not have an importance related to core damage, the team selected the standby gas treatment system based on its importance as a mitigation system in maintaining secondary containment.

The NRC inspectors determined that the team had appropriately considered the importance of the selected systems to risk, prior systematic evaluations, and the systems' operating experience. The selected systems were important for preventing core damage or maintaining secondary containment and they had not been recently reviewed through some other formal inspection effort. The team also selected one system based on a recent history of operating problems. The NRC inspectors concluded that the team had made appropriate choices for the self assessment.

The NRC inspectors also noted that the scope of the team's self-assessment effort was flexible, in that they had expanded the scope of their self assessment as circumstances warranted. For example, the self-assessment team reviewed a reactor feedwater pump runback that occurred during the assessment. They also included an evaluation of the design adequacy of a safety relief valve installed in the standby service water system, after noting that the valve routinely opened following a pump start. This flexibility of the self-assessment team was noted as a strength.

3 SELF-ASSESSMENT FINDINGS AND RELATED CORRECTIVE ACTIONS

3.1 Self-Assessment Conclusions

In a report dated January 29, 1996, the self-assessment team concluded that no significant deficiencies were identified. However, some of the findings were identified as quality deficiencies. These deficiencies involved:

- A long-standing plant deficiency, whose lack of resolution resulted in a challenge to plant safety systems.
- An equipment material condition that caused the inadvertent actuation of a control room alarm.
- One example of a missing design change review for a modification change package.
- An electrical drawing error caused by a design engineer's oversight.
- The failure to update vendor documentation and calculations following equipment replacement.
- Minor differences between the licensee's preferred source of system information (i.e., lesson plans) and the licensing and design basis documents.

The licensee indicated that the self-assessment team deficiencies would be resolved as part of the licensee's corrective action program. The deficiencies are discussed in more detail in Section 3.2 of this report.

The self-assessment team also identified several issues involving the licensee's current engineering and corrective action processes, conduct of engineering, plant material condition and reliability, engineering procedures and documentation, engineering staff knowledge and performance, engineering organization and administration, and the effectiveness of the licensee's self-assessment activities. The self-assessment team concluded that the engineering program was effective in supporting plant operational activities. The self-assessment team's findings and conclusions regarding these issues are discussed in more detail in Sections 3.3 - 3.8 of this report. These issues involved:

- A questionable licensee implemented surveillance test procedure for venting the residual heat removal system. The procedure originally implemented the Technical Specification venting requirements that were based upon expected air intrusion into the system. However, current air intrusion phenomena required the implementation of additional venting to fully vent the system. As the result, the licensee revised the monthly surveillance test procedure to include opening of the auxiliary building local high point vent valves on a 31-day frequency (Section 3.3.1).

- A lack of proactivity by system engineers which contributed to a pump's failure. The self-assessment team concluded that this demonstrated a need for plant management to emphasize aggressiveness in taking proactive steps to prevent component failure (Section 3.3.2).

- An assessment that engineering was accomplishing the valve stem movement check on the licensee's reactor core isolation cooling Terry turbine governor valve stem. This check was recommended as the result of industry experiences regarding potential problems with governor valve stems (Section 3.3.3).

- The identification that the standby service water system pump discharge header relief valve had incorrect information on the nameplate and drawings, as well as, an incorrect valve sizing calculation (Section 3.3.4).

- The observation that plant personnel operated the plant service water system in the manual mode because of concerns over the design adequacy of the automatic controls (Section 3.3.5).

- An assessment that the plant was in good material condition and showed good housekeeping practices in areas routinely accessed, however, less attention to detail was evident in the more obscure areas of the plant (Section 3.4).

- The identification of documentation discrepancies, which indicated a need for a heightened awareness for attention to detail when dealing with original design documentation (Section 3.5).

- An assessment that the design organization was effective in supporting the plant by assuring the atmospheric depressurization system was capable of performing its safety function (Section 3.6).

- An assessment that the licensee's backlog management of various open item tracking systems, such as the nuclear plant engineering and system engineering tracking systems were cumbersome but effective (Section 3.7.1).

- The identification of inconsistencies between the system engineer system handbooks and the management guidelines provided to the system engineers regarding the preparation and use of these handbooks. However, the use of these handbooks was considered to be effective (Section 3.7.3).

- An assessment that the licensee's processes for identifying, resolving, and preventing problems provided for timely identification, determined the cause, and provided appropriate corrective actions to prevent recurrence. However, the licensee's corrective action program did not appear to be as effective in identifying problems that were potential precursors to new problems. In addition, the trending of open significant items was weak and longer term actions received lesser focus (Section 3.8.2).

- The identification that the corrective action process was cumbersome and would benefit from improvements. These improvements included: (1) improvement of root-cause timeliness and increased line organization involvement in root-cause analyses; (2) reduction of the number of corrective action processes; (3) verification of corrective action effectiveness; and, (4) consistent management expectations for reporting thresholds, tracking and closure mechanisms, and meeting due dates (Section 3.8.2).

- The identification of a growing backlog trend primarily due to limited staffing in the root-cause analysis group (Section 3.8.4).

- An assessment that the operating experience program evaluations and dissemination of information to the plant staff of industry experiences was effective (Section 3.8.5).

Based on sample inspections and interviews, the NRC inspectors determined that the self-assessment team's conclusions were, for the most part, appropriate. The self-assessment team noted that no safety degradations or departures from regulatory requirements were identified in the course of the assessment. In contrast to the team's conclusions, the NRC inspectors concluded that minor departures from regulatory requirements occurred. However, the NRC inspectors also concluded that the team's self-assessment had effectively provided findings to licensee management for their evaluation. The NRC inspectors noted that some of the team's conclusions involved enhancements which exceeded regulatory requirements. For example, the NRC inspectors agreed with the self-assessment team's conclusion that while the corrective action and the root-cause analysis programs were effective and met regulatory requirements, these programs could be enhanced. The NRC inspectors also agreed that the operating experience program was effective by providing timely dissemination of information throughout the facility and the industry.

3.2 Licensee-Identified Quality Deficiencies

As a result of the self-assessment team's review, the licensee initiated quality deficiency reports identifying six deficiencies. The NRC inspectors reviewed the deficiencies identified by the self-assessment team to evaluate the safety significance of the issues involved, assess the thoroughness of the team's self assessment, and determine whether violations of NRC requirements occurred. The NRC inspectors concluded that the deficiencies identified did not represent operability concerns or major weaknesses in the engineering or corrective action processes. However, some of the deficiencies by the self-assessment team were minor violations of minimal safety significance. These deficiencies are described in more detail below.

Blocked Open Fire Door Resulted In Unit Downpower (QDR 0007-96)

During the self assessment, a reactor feedwater pump tripped and the operators lowered reactor power to 70 percent. Although the reactor feedwater system was not originally included in the self-assessment scope, the self-assessment team expanded the focus of the assessment to include this event. They interviewed the system engineer and determined that poor ventilation caused elevated room temperatures, which adversely affected the electronic equipment located in the room. Therefore, the licensee had blocked the doors open to keep the room cool. However, when outside temperatures dropped below freezing, an instrument line froze causing the reactor feedwater pump to trip. Licensee personnel closed the door and planned a long-term solution to this problem prior to the onset of summer. These corrective actions were not completed prior to the end of the team's self assessment.

The self-assessment team concluded that this was an example of a long-standing plant deficiency, which was not properly recognized and whose lack of resolution resulted in a challenge to plant safety systems.

Although blocking the doors open was an example of a poor corrective action, the NRC inspectors determined that a violation of NRC requirements did not occur. The NRC inspectors noted that the original adverse condition was corrected in a way which would prevent recurrence (i.e., during hot weather the door was blocked open allowing sufficient cooling flow to the electronic equipment). The NRC inspectors agreed with the self-assessment team's conclusions that the corrective action was weak because licensee personnel did not fully evaluate the long-term implications of operating with the doors blocked open.

Spurious Reactor Core Isolation Cooling Pump Room Flood Alarm Actuation (QDR 0008-96)

During a walkdown of the reactor core isolation cooling pump room, the self-assessment team noticed that a cable extended a few inches from an open pipe into the walkway. The team members bumped the cable while examining it, which inadvertently actuated a room flooding alarm. The licensee later determined that the cable, which was at the bottom of a float located inside the pipe, should have been cut off. The licensee trimmed the cable and initiated the quality deficiency report.

The quality deficiency report was written to document the false alarm actuation. If this alarm represented an actual flooding condition, the licensee would have entered an emergency response procedure. The intent of the quality deficiency report was to heighten personnel awareness of the event, the sensitivity of the room flooding level switch, and the potential for inadvertent actuation.

During their plant tours, the self-assessment team did not observe any other similar conditions in the plant. As a result, the team concluded that this event was an isolated case. The NRC inspectors concurred with the self-assessment team's conclusion and further concluded that this event represented a very minor concern that was properly resolved.

Lack of Design Change Review By the Motor-Operated Valve Engineer (QDR 0010-96)

During the last refueling outage, the licensee increased the size of the minimum flow line restricting orifice for the reactor core isolation cooling system. During review of this modification, the self-assessment team noted that licensee personnel had not considered the effect of the flow increase on a motor-operated valve located downstream of the orifice.

As a result of the team's finding, licensee personnel evaluated the effect of increasing the flow orifice size on the motor-operated valve and determined that the installed modification was acceptable. Licensee personnel also stated that there was a failure to involve the motor-operated valve experts in the review of the design. They indicated that appropriate personnel are trained to ensure that the motor-operated valve experts review applicable design changes. However, in this instance, an oversight occurred and the review was not accomplished. The NRC inspectors concluded that this was an isolated occurrence because there was no other evidence that such reviews had been missed.

Residual Heat Removal Pump Electrical Drawing Not Correctly Annotated to Address Train Differences (QDR 0011-96)

The self-assessment team reviewed Modification Change Package 92-1049, which changed a Residual Heat Removal Pump A control circuit relay to a time delay relay. The team noted that the electrical control circuit drawing was applicable to both Pump A and Pump B. Since the design engineer did not specify that the change was only applicable to Pump A, the revised drawing incorrectly indicated that a time delay relay was also installed in the control circuit for Pump B. The self-assessment team noted that this was the only error detected during their review of a large number of drawings. On that basis the team concluded that the error was isolated occurrence. The NRC inspectors concluded that the sampling of a large number of drawings with negative similar findings was a reasonable basis for considering this error to be an isolated occurrence.

Criterion III of 10 CFR 50, Appendix B, requires that design changes be subject to design control measures, which are commensurate with those applied to the original design. The NRC inspectors noted that the failure to correctly update the electrical control drawing constitutes a violation of minor significance, which is being treated as a Non-Cited Violation, consistent with Section IV of the NRC Enforcement Policy.

Vendor Documentation and Calculation Not Updated Following Replacement of 125V DC Automatic Depressurization System Solenoid Coils (QDR 0012-96)

The self-assessment team reviewed Design Change Package 84-3107, which replaced the two solenoids on each of the 20 safety-relief valves to meet the service environment conditions inside the drywell. The self-assessment team identified that the licensee had not updated the vendor documentation and the associated battery sizing calculation when the solenoid type changed. The self-assessment team further determined that the change in amperage requirements for the new solenoid valve (0.18 amps for the new solenoid versus 0.16 amps used in the battery size calculation) had no affect on the end result and did not change the pass/fail criterion of the calculation. The self-assessment team did not identify any similar cases during their review of other design change packages. The NRC inspectors concluded that the team's review efforts were appropriate and complete.

Criterion III of 10 CFR 50, Appendix B, requires that design changes be subject to design control measures which are commensurate with those applied to the original design. The NRC inspectors noted that the failure to correctly update the battery sizing calculation constitutes a violation of minor significance, which is being treated as a Non-Cited violation, consistent with Section IV of the NRC Enforcement Policy.

Lesson Plan Figure Not Updated (QDR 0018-96)

The self-assessment team reviewed Lesson Plan OP-LO-SYS-LP-E22-2-04, "Automatic Depressurization System/Safety Relief Valves," to determine if it was consistent with the licensing and design basis documentation. The team determined that lesson plans were viewed by the licensee as the preferred source of general system information. The licensee planned to use the lesson plans to supersede the system description manual. The self-assessment team identified minor differences between the lesson plan and the licensing and design basis documents. They also found that Figure 2A of the lesson plan was not revised to reflect the deletion of check valves and relief valves in Design Change Package 90/0005, even though Figure 2B showed the system pictorially correct. The self-assessment team concluded that the updating process for the lesson plans was not effective. This conclusion was evidenced by the fact that engineering did not review the lesson plans and that there were interim changes that needed to be incorporated.

The NRC inspectors agreed with the self-assessment team that this failure to update the lesson plan figure was a weakness and evidence that the updating of the lesson plans was less than effective. The NRC inspectors also noted that lesson plans were not design control documents and, therefore, the requirements of 10 CFR Part 50, Appendix B, Criterion III did not apply.

3.3 Conduct of Engineering

The self-assessment team reviewed the conduct of engineering through personnel interviews and document reviews. The team concluded that both system and design engineering were performed in a manner that enhanced the safety and reliability of the plant. However, the team identified several minor instances where weaknesses in the conduct of the engineering activities were observed. These weaknesses included:

- A lack of proactive response to the residual heat removal system air intrusion event and jockey pump failure;
- A lack of coordination between site and corporate engineering with respect to the replacement of the reactor core isolation cooling pump turbine governor valve stem;
- The acceptance of a poor practice regarding lifting of a relief valve during standby service water pump startup; and
- A lack of resolution of a long standing issue regarding the operation of the plant service water in the manual mode.

These instances are described more fully in the following sections.

3.3.1 Residual Heat Removal System Train B Piping Air Intrusion

The self-assessment team noted that the licensee had a long-standing issue related to air entrainment in Train B of the residual heat removal system. In 1991, the licensee first identified abnormal pulsating noises in the area of high point vent valves (1E12F321 and 1E12F351) on the test return line. Over the years, the licensee had performed a number of activities to determine the cause of the intermittent noises. The licensee had performed visual inspections, venting operations, acoustical monitoring, ultrasonic testing, and employed divers to determine the cause of the noises. During this period, the licensee theorized that the noises were caused by air introduced during operation of the suppression pool cleanup system.

When the abnormal noises reappeared in January 1995, the licensee noted that air vented from Valves 1E12F400B and 1E12F401B. These valves were located near the residual heat removal heat exchanger and provided the local high point vent for the residual heat removal system piping located in the auxiliary building. As the result of this finding, licensee personnel began implementing a venting procedure of the system every shift to ensure that Train B of the residual heat removal system remained free of voids. At that time, the licensee also noted that air was not observed when the containment building high point vent valves (1E12F107 and 1E12F108) were opened.

Technical Specification Surveillance 3.5.1.1 required that the residual heat removal discharge piping be vented every 31 days. The basis for this venting was operating experience and the gradual nature of void buildup in the residual heat removal discharge piping. Licensee personnel developed a procedure to implement this surveillance requirement. This procedure directed personnel to open the containment building high point vent valves (1E12F107 and 1E12F108) on a 31-day frequency, but did not include the auxiliary building local high point vent valves (1E12F400B and 1E12F401B).

While onsite, the self-assessment team questioned the adequacy of the surveillance test procedure to implement the Technical Specification basis given the current air intrusion phenomena and the additional venting required to fully vent the system. As the result of the team's questions, the licensee revised the monthly surveillance test procedure to include opening the auxiliary building local high point vent valves on a 31-day frequency.

During the second week of the NRC inspection, licensee personnel identified that the air in-leakage was actually caused by steam leaking into the residual heat removal system from abandoned cross-connect steam header piping for the reactor core isolation cooling system. The licensee repaired the leaking valves and concluded that the original theories about air intrusion from the suppression pool clean up system were not valid.

While licensee personnel did revise the surveillance test procedure in response to the self-assessment team's questioning, the NRC inspectors noted that engineering was not proactive in that the surveillance test procedure was not revised when it first became apparent that the basis for the surveillance requirement was suspect. The NRC inspectors considered the lack of proactivity to be the causal factor for the less than timely revision to the surveillance procedure. The NRC inspectors concluded that while the change was not timely, the change to the surveillance procedure was not needed to support system operability. The NRC inspectors also concluded that the self-assessment team was influential in focusing the licensee's attention toward resolving this issue.

3.3.2 Residual Heat Removal Jockey Pump Failure

The self-assessment team identified that a system engineer did not aggressively respond to increasing bearing temperature and vibration data on the residual heat removal system jockey pump. The team had reviewed Material Nonconformance Report 0245-95, which identified the residual heat removal Train B jockey pump failure. In addition, they reviewed the E-12 system quarterly reports for the fourth quarter of 1994 and the first, second, and third quarters of 1995; and, the jockey pump bearing temperature and vibration trends from May 1992 to October 1995. The self-assessment team noted the increase in both jockey pump bearing temperature and bearing vibration over the months following the jockey pump replacement in February 1995 until the pump failed in August 1995. While the trend of increased bearing and vibration temperature was rapid, the increased bearing vibration and temperature levels were still below the inservice test acceptance levels during this time period.

In early August 1995, the system engineer obtained temperature and vibration data during his weekly walkdown, while noting that the pump sounded different than on previous walkdowns. Analysis of this vibration data by the vibration engineer did not suggest pump failure was imminent. Further research into the issue put the setup of the jockey pump's oiler in question. The oiler was adjusted and the jockey pump failed 3 days after this adjustment was completed. The self-assessment team noted that while licensee personnel attempted to be proactive, their approach was ineffective in preventing pump failure. The self-assessment team concluded that this demonstrated a need for plant management to emphasize aggressiveness in taking proactive steps to prevent component failure.

Through additional interviews, the self-assessment team determined that many system engineers felt they are consumed with daily work demands leaving little time for nonimmediate concerns.

The NRC inspectors noted the team's findings and observations and agreed with their conclusion. The NRC inspectors also noted that while the system engineers were busy with their daily work demands, there was progress toward making the system engineers more proactive at identifying system problems.

3.3.3 Corrosion Susceptible Reactor Core Isolation Cooling Pump Turbine Governor Valve Stem

During the fall of 1995, Region IV conducted an initiative in response to increasing failures of Terry turbines within the industry. The purpose of this initiative was to determine the effectiveness of the licensee's response to this industry experience information. While this initiative did not include the Grand Gulf Nuclear Station facility, the NRC forwarded a copy of a finalized inspection report to the licensee for their information.

Based upon their knowledge of this previous NRC initiative, the self-assessment team conducted a review of the reactor core isolation cooling pump, which is driven by a Terry turbine, to determine the licensee's actions with respect to this initiative. The self-assessment team noted during their walkdown of the reactor core isolation cooling pump room that there was condensate dripping at the rate of about 1 drop per minute from the governor valve stem. The team questioned the system engineer regarding this leakage to determine the stem material. The system engineer stated that this valve stem had recently been replaced (approximately 6 months ago). The engineer believed that the replacement stem was manufactured from material which was susceptible to corrosion in a moist environment.

The self-assessment team determined that the licensee had formed a specific team to evaluate the applicability of the Terry turbine initiative results to the Grand Gulf Nuclear Station. Since this specific team's evaluation was in progress during the self-assessment team's visit, the self-assessment team did not fully assess the licensee in this area. However, the self-assessment team concluded that the licensee's on-going evaluation needed to review other areas identified from this initiative such as steam admission valve leakage, low point drains, governor valve stem material, and governor valve maintenance frequency.

The NRC inspectors concluded that the self-assessment team appropriately deferred full evaluation of the Terry turbine until after licensee personnel had an opportunity to evaluate the results of the Terry turbine initiative. The self-assessment team was aware of the important aspects of the industry initiative and they had highlighted these issues to the licensee for their consideration in their on-going evaluation.

The NRC inspectors accompanied the self-assessment team during their walkdown of the reactor core cooling pump room with the system engineer. During the second week of the inspection, the NRC inspectors again toured the reactor core isolation cooling pump room with the system engineer. The NRC inspectors determined that the system engineer was knowledgeable of the system.

During this second week, the system engineer also provided the NRC inspectors with a final draft of the licensee's assessment of the applicability of the Region IV's Terry turbine initiative findings, which was not available to the self-assessment team. The NRC inspectors determined that with one exception the identified industry issues had been addressed. The exception involved the replacement of the governor valve stem. The NRC inspectors noted that despite failures at other Entergy Operations facilities, licensee personnel had no firm plans to inspect or replace the installed corrosion susceptible reactor core isolation cooling pump turbine governor valve stem during the next outage.

Based upon the evidence of water dripping from the governor valve stem, that was identified by the self-assessment team, the NRC inspectors concluded that the valve stem was in a moist environment. In lieu of valve stem replacement, the licensee was manipulating the governor valve on a weekly basis to detect binding. The NRC inspectors noted that the Region IV initiative had previously identified that corrosion of this valve stem was the cause of mechanical overspeed trips of Terry turbines at other Entergy Operations facilities. While the weekly valve manipulation provided assurance that the governor valve was operable during the time period between normal tests, personnel at these other facilities had determined that the corrosion mechanism can be rapid and that governor valve manipulation may not detect valve stem degradation prior to failure.

The NRC inspectors concluded that a lack of consistency between the Entergy Operations engineering organizations existed. The NRC inspectors also concluded that this lack of consistency resulted in a nonconservative corrective action plan for governor valve stem degradation.

3.3.4 Standby Service Water System Relief Valve Operation

During plant tours, the self-assessment team noted that the standby service water system pump discharge header relief valve (1P41F299B) lifted when the pump started and reseated shortly after flow was established. Even though this system was not one of the selected systems, this observation prompted the self-assessment team to expand their review to include this system. The team reviewed the valve's nameplate, drawings, and sizing Calculation 2.2.40. Although the installation was acceptable, the self-assessment team reported that the sizing basis calculation appeared to be incorrect and that the valve nameplate was incorrect. In addition, the self-assessment team also considered it to be a poor practice to use a safety relief valve to provide minimum flow recirculation for the pump.

The licensee acknowledged the self-assessment team's observations and entered these observations into their corrective action program for review and resolution. The licensee's corrective actions were not completed prior to the conclusion of the self assessment.

The NRC inspectors performed a followup inspection of the licensee's corrective actions taken to resolve the team's findings. From this review the NRC inspectors determined that the valve sizing calculation in the design basis documentation was correct. The NRC inspectors did not identify any further errors in the sizing calculation and determined that even if the valve were to fail open, system operability would not be affected. However, the NRC inspectors were in agreement with the self-assessment team's finding that it was a poor practice to use a safety relief valve to provide minimum flow recirculation for the pump. The NRC inspectors also noted the relief valve nameplate error identified by the self-assessment team. Licensee personnel indicated that correct tags were supposed to be installed as a part of a 1983 modification that changed the valves from steam valves to liquid valves. The licensee further speculated that the tags may have been removed during painting operations. The licensee stated that the correct tags would be installed.

3.3.5 Plant Service Water Radial Pump Well Operation

The radial well pumps for the plant service water system (a nonsafety-related system) have both manual and automatic controls. In automatic, the pumps can operate either in pressure control or flow control. The purpose of the automatic controls is to adjust pump output to the changing needs of the plant or to compensate for other pump problems. The self-assessment team noted that plant personnel operated the plant service water system in the manual mode because of concerns over the design adequacy of the automatic controls. The self-assessment team was concerned that if plant service water loads automatically increased, the system resistance would decrease. As a result, with the pump in manual, pump output could increase too far and the well may pump down or the motor current may be exceeded and cause a pump trip just as the well was most needed.

The self-assessment team determined that operating these pumps in the manual mode introduced additional reliability challenges. They noted one event where plant personnel had to reduce plant power to compensate for a plant service water system malfunction. The self-assessment team reported that other plant service water initiated events had occurred that affected the plant to a lesser degree. One of the control room annunciators for the system was also noted to be out of service. As a result, the first indication that the plant service water system was not functioning properly would be indication of changing plant conditions caused by the plant service water system malfunction. Based upon the self-assessment team's findings, the licensee will review the operation of the plant service water system to determine the cost/benefit of redesigning the system's controls such that reliable automatic operations can be obtained. The self-assessment team did not identify any other examples of systems operating in manual due to inadequate automatic control systems.

The self-assessment team also noted that the system engineer monitored trends of well performance and that plant personnel were focused on improving well performance. Licensee personnel had identified that Well #1 was an operator work-around, due to poor performance. Work completed on other wells had significantly improved their performance and plans were being made to clean Well #1.

The NRC inspectors concurred with the self-assessment team's findings. In addition, the NRC inspectors considered consistently operating in the manual mode of operation to be an operator work-around. The NRC inspectors noted through review of past events and interviews, that this type of operation leaves the plant in a reactive vice a proactive mode of operation with respect to plant service water system operation.

3.4 Plant Material Condition and Equipment Reliability

The self-assessment team performed a general plant walkdown and walkdowns of the specific systems selected for review during the assessment. In general, the team concluded that the plant was in good material condition and showed good housekeeping practices in areas routinely accessed. Some of the findings identified by the team included:

- A remote camera monitor cart was chained to a fire hose reel and had the potential to interfere with the hose reel use. The licensee determined that this configuration did not make the hose reel inoperable, but moved the cart to ensure that an interference did not occur.

- A barricade was erected to control entry and an antistatic mat was installed in front of the automatic depresurization system control panel area in the control room. The purpose of these items was to prevent the inadvertent actuation of the automatic depresurization system which had been a problem in the past. As the result of subsequent review, these items were considered to be appropriate by the team.

- Temporary lead shielding was installed on shutdown cooling system lines. Subsequent investigation by the licensee provided evidence that the weight of the shielding had been properly evaluated for seismic loading concerns.

- A string was observed to be coming from the reactor core isolation cooling room flooding level switch. See Section 3.2 of this report for further information on this finding.

- A 55 gallon drum lid was noted to be hanging from a valve flange. The team considered this configuration to be inconsistent with plant design drawings. However, further review by the licensee determined that the configuration was correct and consistent with plant drawings

- A burned out light was observed in the reactor core isolation cooling system room. The licensee initiated actions to replace the light bulb.

- A drip was noted from the governor valve stem on the reactor core isolation cooling system pump turbine and an elevated temperature was noted in this room. See Section 3.3.3 of this report regarding further information on the drip from the governor valve stem. The licensee determined that the room's elevated temperature had been reviewed and was acceptable.

- The existence of an incorrect nameplate and the observed lifting of the relief valve on the standby service water pump. See Section 3.3.4 of this report for further information regarding this finding.

- The observation that the plant service water pumps were being operated in the manual mode of operation. See Section 3.3.5 of this report regarding further information regarding operation in this mode.

In addition, the team noted that less attention to detail was evident in the more obscure areas of the plant.

The NRC inspectors toured the plant to verify the self-assessment team's findings. The NRC inspectors also noted that areas that were heavily trafficked were in very good condition, however, areas not easily accessed or trafficked were considered to be only in an adequate condition. Poorer areas included the reactor water cleanup sample sink area, the drywell purge compressors, the hydraulic control unit area, the reactor feedwater pump area, and the radial pump wells. The NRC inspector's positive observations included a very clear suppression pool, an absence of system leaks, and very good conditions for the standby liquid control system, high pressure core spray system, reactor core isolation cooling system, and the emergency diesel generators.

To further measure equipment condition, the NRC inspectors requested reliability data for the reactor core isolation cooling system and the residual heat removal system. The NRC inspectors reviewed this data and noted that safety system reliability for each of these systems for the past 2 years considerably exceeded the licensee's goals.

The NRC inspectors also questioned operations personnel about the effectiveness of the engineering support that they received. From this questioning, the NRC inspectors concluded that operations personnel were very satisfied with the support they received from the engineering organizations. This finding was consistent with the findings that the self-assessment team had in this area.

3.5 Engineering Procedures and Documentation

The self-assessment team identified several minor discrepancies between licensing basis documents, drawings, calculations, and training manuals. Licensee personnel subsequently determined that in each case, the design met the safety functional requirements. The self-assessment team noted that the licensee used station training manuals to capture system description information. The licensee planned to correct the system documentation, including making clarifications to the Updated Final Safety Analysis Report.

As an example, the self-assessment team identified that the pneumatic supply accumulators for the automatic depressurization system were not sized to operate the automatic depressurization system valves against 70 percent of drywell design gage pressure (21.0 psig) as documented in Section 7.3.1.1.1.4.2 of the Updated Final Safety Analysis Report. The accumulators were sized to operate the automatic depressurization system valves against 70 percent of drywell design absolute pressure (31.3 psia). As a result, it appeared that the sizing for the accumulators was nonconservative by 4.4 psig with respect to the Updated Safety Analysis Report.

Licensee personnel concurred with this observation, but subsequently determined that the additional 4.4 psi was not warranted. They stated that, for intermediate and small break accidents, drywell pressure would not exceed 16 psig (30.7 psia) at the time the automatic depressurization system would be actuated. They further stated that the accumulators were actually sized with a 0.6 psi margin with respect to a predicted drywell pressure of 30.7 psia. This conclusion was consistent with later licensing basis documentation that supported the use of 70 percent of drywell design absolute pressure versus 70 percent of drywell design gage pressure. The NRC inspectors noted that licensee personnel had initiated Licensing Document Change Request 96-0011 to correct the Updated Final Safety Analysis Report.

Licensee personnel were required by 10 CFR 50.71(e) to periodically update the Final Safety Analysis Report to assure that the information was current. The failure to update the Final Safety Analysis Report as the analysis evolved, constituted a violation of minor significance, which is being treated as a Non-Cited Violation, consistent with Section IV of the NRC Enforcement Policy.

The NRC inspectors agreed with the assessment team's conclusion that the documentation discrepancies identified by the team indicated a need for a heightened awareness for attention to detail when dealing with original design documentation.

3.6 Engineering Staff Knowledge and Performance

As stated in Section 2.3 of this report, the self-assessment team used a vertical review and broad review approach to evaluate the capabilities and performance of the engineering organizations. The team's vertical review evaluation of the automatic depressurization system was accomplished by conducting a series of interviews with the associated engineers, walking down accessible sections of the systems in the plant, and by performing detailed reviews of drawings, design change packages, nonconformance reports, calculations, and engineering reports. The self-assessment team concluded that the design organization was effective in supporting the plant by assuring the atmospheric depressurization system was capable of performing its safety function.

The team's broad review of the residual heat removal, standby gas treatment, reactor core isolation cooling, and plant service water systems found that system engineers were knowledgeable in many aspects of their systems and were knowledgeable of open items related to their systems. The system engineers demonstrated ownership of their systems and provided good support at all of the system outage planning meetings.

3.7 Engineering Organization and Administration

3.7.1 Backlog Management

The self-assessment team conducted reviews of the licensee's backlog management of various open item tracking systems. The team noted that there were instances where the process was cumbersome (e.g., the nuclear plant engineering and system engineering tracking systems), however, the licensee was able to provide all requested information. The team also noted that backlogs, except for outstanding corrective actions, were routinely identified in monthly trend reports. While this report provided information regarding the average age of the outstanding issues, the reason for the age of the item, the significance of the issue, or a trend of the item status was not addressed.

As an independent review of the backlog management, the NRC inspectors requested and received historical information regarding backlogs for quality deficiency reports, engineering evaluation requests, and material nonconformance reports. The NRC inspectors compared the number of quality deficiency reports currently open (102) with the number of quality deficiency reports open during a similar time frame in the last refueling cycle (143) and noted that the backlog had a downward trend. A similar evaluation of open engineering evaluation requests and open material nonconformance reports was performed. The number of open engineering evaluation requests and material nonconformance reports was not excessive, but had not changed significantly. The number of open material nonconformance reports had trended down slightly and the number of open engineering evaluation requests had trended up slightly. The NRC inspectors concluded that the licensee's efforts to manage completion of corrective actions was effective.

The NRC inspectors also reviewed selected open engineering evaluation requests to determine if there were any open Priority 1 and 2 engineering evaluation requests. Consistent with the self-assessment team findings, the NRC inspectors did not identify any Priority 1 open engineering evaluation requests. The NRC inspectors reviewed open Priority 2 engineering evaluation requests to determine if action and timeliness on these engineering evaluation requests was appropriate. No problems were identified.

The NRC inspectors concluded that licensee personnel had established systems for effectively managing work backlogs. This was evidenced by the low backlogs and generally decreasing trends.

3.7.2 Review of Cancelled Modification Requests

An area not reviewed by the self-assessment team was the basis for cancelled modification requests. A review of such requests can provide information for determining if any safety significant modifications were inappropriately canceled. This provided another check on the quality of the engineering activities. As a check on this activity, the NRC inspectors reviewed a listing of modifications cancelled within the past two years. This provided another check on the quality of the engineering activities. The NRC inspectors noted that a small number (58) were cancelled and that no problems were identified with these cancellations. The NRC inspectors concluded that all cancelled modification requests were appropriate and did not adversely affect plant safety systems.

3.7.3 Use of System Engineering Handbooks

The self-assessment team considered the system engineering handbooks to be excellent documents that clearly conveyed management's expectations. However, the self-assessment team noted some inconsistencies between the system engineer's maintenance and use of these handbooks and the management guidelines.

When the team discussed this observation with licensee management personnel, they were informed that the system engineer preparation and use of the handbooks was consistent with their expectations. While two of the team members considered more consistency with the guidelines to be an enhancement item, the team concluded that the use of the handbooks was effective.

The NRC inspectors interviewed the system engineers regarding the use of the system engineering handbooks and reviewed selected handbooks. The NRC inspectors agreed with the team that the handbooks were an effective and useful tool and concluded that the system notebooks were useful documents that provided important system information.

3.8 Effectiveness of Self-Assessment Activities and Corrective Action Processes

The self-assessment team performed an assessment of the licensee's controls in identifying, resolving, and preventing problems. These controls included a review of the safety review committees, root-cause analysis program, corrective action program, self-assessment process, and processes that provide for the incorporation of operating experience feedback.

3.8.1 Self-Assessment Team Performance

The self-assessment team used previous self assessments as a basis for some of the findings and conclusions. For example, the self-assessment team relied on information in previous licensee self assessments of the 10 CFR 50.59 process, engineering, corrective action, employee concerns, and previous plant trips. Licensee personnel conducted these previous self assessments during the period of August through December 1995.

The NRC inspectors reviewed these prior self assessments and determined that the use of these self assessments was appropriate and provided additional insights. As previously indicated, the NRC inspectors determined that the self-assessment team was very effective in conducting this self assessment. The self-assessment team members conducted in-depth reviews based upon their experience at their own plants (for Entergy Operations personnel) or their facilities. As a result, the self-assessment team provided a fresh and independent assessment of Grand Gulf Nuclear Station.

3.8.2 Corrective Action Process

The self-assessment team determined that the processes for identifying, resolving, and preventing problems provided for timely identification, determined the cause, and provided appropriate corrective action to prevent recurrence. The team also noted that the issues were being tracked and trended. The self-assessment team noted that the corrective action process was identifying approximately 1100 issues per year with a monthly average of 65 issues during months the plant was in operation and 190 issues during outage periods. They considered these numbers to be consistent with numbers observed at other facilities. However, the team also noted that the problems identified were known problems (i.e., they did not appear to be effective in identifying problems that were precursors to new problems).

The team also noted that the licensee used a monthly trending report and an annunciator window report to both track and trend information. While these reports were effective for tracking and trending information, the team determined that the trending of open significant items was weak and that, while immediate or short-term items were aggressively pursued, longer term actions received lesser focus. The team also determined that the number of overdue corrective actions was small.

While the self-assessment team did not view the corrective action processes as deficient, there was evidence that the process was cumbersome and would benefit from improvements. A December 1995 self assessment of the corrective action program identified similar findings. As a result of the December self assessment, a quality deficiency report was issued to track corrective action program improvements. This quality deficiency report identified the following items as areas for potential improvement:

- Improve root-cause timeliness and increase line organization involvement in root-cause analyses;
- Reduce the number of corrective action processes;
- Provide for verification of corrective action effectiveness; and,
- Provide consistent management expectations with respect to reporting thresholds, tracking and closure mechanisms, and meeting due dates.

The self-assessment team concluded that these were significant, fundamental improvements which should improve performance. They also noted that the licensee was making progress toward implementing a new corrective action program.

Due to the relative recent issuance of the self-assessment report, the NRC inspectors noted that a number of minor followup items were not yet entered into a corrective action tracking system. Licensee personnel indicated that their reviews to identify and classify items, was ongoing and not yet complete. The licensee planned to update the performance data system and the engineering tracking systems such that these followup items receive appropriate tracking and trending. Based upon interviews and reviews of other self-assessment results, it was evident that these other self-assessment findings were properly entered into the licensee's various tracking systems for resolution. The NRC inspectors concluded that there was high confidence that the findings from this self assessment would receive appropriate attention.

3.8.3 Top 10 Quality Issues List

The self-assessment team reviewed the methods used by the licensee to track repetitive issues. The team noted that such issues were monitored and processed via the monthly trend reports that used information from the performance data system. From this data, a "Top 10 Quality Issues" list was formulated. This list was than reviewed to determine the reason for the repetitive activity and to develop methods to prevent further repetition. The self-assessment team concluded that the Top 10 Quality Issues list was a highly useful document that enhanced the corrective action process.

The NRC inspectors agreed with the team's conclusion and considered the use of the Top 10 quality issue list to identify and track repetitive issues to be effective for assuring that such issues are resolved in a timely manner.

3.8.4 Root-Cause Analysis Program

The self-assessment team reviewed the licensee's root-cause analysis program to determine the effectiveness of the program and to determine the root-cause analysis backlog. The team noted that while root-cause analysis was a sound program and that root-cause analyses were being appropriately requested, that there was a growing backlog trend primarily due to a limited staffing in the root-cause group. The team noted that if the initiatives identified in the December 1995, self assessment of the corrective action program (which addressed the root-cause program) were implemented, these backlogs could be reduced. In addition, the team considered that an increased use of incident review boards would assure that root-cause analyses would be initiated in a more timely manner.

Further issues identified by the team included the existence of root-cause analyses which were either too broad or too narrowly focused. Those analyses that were focused too narrowly resulted in the underlying causes to be missed and contributed to the occurrence of repeat events. The team also noted the recent establishment of a corrective action review board. The team considered this to be an important improvement toward providing more consistency and a better management overview of the corrective action program.

The NRC inspectors concurred with the self-assessment team's findings and observed the activities of the corrective action review board. The NRC inspectors considered the corrective action review board to be an enhancement to the root-cause analysis program.

3.8.5 Review of Operating Experience Feedback

The adequacy of the licensee's operating experience program was reviewed by the self-assessment team to determine the effectiveness of the evaluations, the dissemination of information to the plant staff and industry, and the corrective actions. The review encompassed documents generated by the Institute of Nuclear Power Operations, the NRC, reports from vendors, and 10 CFR Part 21 notifications. The self-assessment team determined that the licensee's program was effective and noted that initial information screenings received two levels of review. The team also determined that the operating experience organization provided comprehensive monthly reports, was properly disseminating information to appropriate site organizations and personnel, and was providing timely information to the industry.

Enhancements to this program recommended by the team included providing access to and tracking of the interim status of open issues and limiting due dates for closure of operating experience documents to those involving safety significant issues.

The NRC inspectors concurred with the team's conclusions and considered the operating experience program to be effective.

4 REVIEW OF UPDATED FINAL SAFETY ANALYSIS REPORT COMMITMENTS

A recent discovery of a licensee operating their facility in a manner contrary to the Updated Final Safety Analysis Report description highlighted the need for a special focused review that compares plant practices, procedures and/or parameters to the Updated Final Safety Analysis Report description. While performing the inspections discussed in this report, the NRC inspectors reviewed the applicable portions of the Updated Final Safety Analysis Report that related to the areas inspected. The following inconsistency was noted between the wording of the Updated Final Safety Analysis Report and design calculations observed by the NRC inspectors.

As described in Section 3.5 of this report, the self-assessment team noted that there were inconsistencies between the description for the automatic depressurization system sizing in the Updated Final Safety Analysis Report and the sizing design calculations.

ATTACHMENT

1 PERSONS CONTACTED

1.1 Licensee Personnel

#R. Powers, Acting Vice President and Plant Manager
#D. Brosnan, Supervisor, Licensing and Design Basis
*#W. Crockett, Manager, Nuclear Quality Services
*R. Curb, Acting Plant Manager
*L. Fisher, Supervisor, Special Projects, Security Section
#T. Grebel, Director, Regulatory Compliance
#L. Hagen, Director, Safety, Health, and Emergency Services
*B. Hansen-Harris, Special Projects, Security Section
K. Hitchen, Westinghouse Site Representative
*J. Hubble, Supervisor, Operations, Security Section
#D. Miklush, Manager, Engineering Services
#J. Molden, Manager, Maintenance Services
*D. Morris, Special Projects, Security Section
*R. Prigmore, Engineer, Quality Assurance
*#W. Ryan, Supervisor, Access and Fitness-for-Duty
*D. Sisk, Sr. Engineer, NRC Regulatory Support
*R. Taylor, Engineer, Quality Assurance
*R. Todaro, Director, Security Section
*R. Watson, Engineer, Quality Assurance
*#J. Young, Director, Quality Assurance

1.2 NRC Personnel

*J. Dixon-Herrity, Resident Inspector

*Denotes those that attended the exit interview on January 12, 1996.

#Denotes those that participated in the followup telephone exit meeting on April 5, 1996.

In addition to the personnel listed above, the inspector contacted other personnel during this inspection period. Those employees included members of the licensee's technical and management staff and members of the security organization.

2 EXIT INTERVIEW

An exit meeting was conducted on January 12, 1996, following the onsite inspection. A followup telephone exit meeting was also conducted on April 5, 1996. During these meetings, the inspector reviewed the scope and findings of the report. The licensee discussed and furnished proprietary information to the inspector during the course of this inspection. During the April 5, 1996, meeting, the licensee acknowledged their understanding of the apparent violation.