UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30323

Report No.: 50-416/90-25

Licensee: Entergy Operations, Inc.
Jackson, MS 39205

Docket No.: 50-416 License No.: NPF-29

Facility Name: Grand Gulf Nuclear Station

Inspection Conducted: November 17 through January 4, 1990

Inspector: [signature] 1/28/91
J. L. Mathis, Senior Resident Inspector — Date Signed

Approved by: [signature] 1/28/91
F. S. Cantrell, Section Chief — Date Signed
Reactor Projects Branch 1
Division of Reactor Projects

SUMMARY

Scope:

The resident inspectors conducted a routine inspection in the following areas: operational safety verification, maintenance observation, surveillance observation, action on previous inspection findings, on-site followup on the B recirculation pump shaft crack, and reportable occurrences. The inspectors conducted backshift inspections on November 17, and December 5, 12, 19, 1990 and January 2, 1991.

Results:

During this inspection period one violation was identified. An operator failed to follow procedures on November 24, 1990, for placing the second RWCU pump in service at the required reactor pressure which resulted in a RWCU isolation, paragraph 3. This violation does not appear programmatic in nature. The manaul scram that was initiated on November 24, 1990, due to control assemblies drifting out of sequence exemplifies poor planning by operations. Taken together it appears that startup activities may have been rushed.

The coordination of management, engineering and maintenance for the change out of the A and B recirculation pump internals was excellent during the recovery operations as indicated by a total down time of 12 days.

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REPORT DETAILS

1. Persons Contacted

Licensee Employees

W. T. Cottle, Vice President, Nuclear Operations
D. G. Cupstid, Manager, Plant Projects
*L. F. Daughtery, Compliance Supervisor
*M. A. Dietrich, Director, Quality Programs
*J. P. Dimmette, Manager, Plant Maintenance
*C. W. Ellsaesser, Operations Superintendent
*C. R. Hutchinson, GGNS General Manager
F. K. Mangan, Director, Plant Projects and Support
*M. J. Meisner, Director, Nuclear Licensing
L. B. Moulder, Acting Manager, Plant Support
*J. V. Parrish, Manager, Plant Operations
*W. R. Patterson, Assist., General Manager
*J. C. Roberts, Manager, Plant & System Engineering
J. E. Reaves, Manager, Quality Services
S. W. Titus, Director, Nuclear Plant Engineering
G. W. Vining, Manager, Plant Modification and Construction
*G. Zinke, Superintendent, Plant Licensing

Other licensee employees contacted included superintendents, supervisors, technicians, operators, security force members, and office personnel.

*Attended exit interview

2. Plant Status

At the beginning of this inspection period the plant was in mode 4, Cold Shutdown, coming out of refueling outage 4. On November 24, 1990, the plant was manually scrammed due to more than 3 control rods drifting out of sequence. On December 10, 1990, the plant scram due to an instrument air pipe rupture which resulted in a low level scram. On December 18, 1990, the plant was shutdown for an unscheluded outage because of high vibration on B recirculation pump. Both A and B pump internals were replaced due to a thru-wall crack on the B shaft. At the end of the inspection period, the plant was at 100 percent power.

3. Operational Safety, (71707)

The inspectors were aware of the overall plant status, and of any significant safety matters related to plant operations. Daily discussions were held with plant management and various members of the plant operating staff. The inspectors made frequent visits to the control room. Observations included: the verification of instrument readings, setpoints and recordings; the review of operating system status and the tagging of equipment; the verification of annunciator alarms, limiting conditions for operation, and temporary alterations; and the review of daily journals, data sheet entries, control room manning, and access controls.

Selected engineered safety feature (ESF) systems were confirmed operable weekly. The inspectors verified that accessible valve flow path alignment was correct, power supply breaker and fuse status was correct and instrumentation was operational. The inspector verified HPCS, LPCS, and CRD systems operable using the probabilistic risk assessment based system inspection plan.

The inspectors conducted plant tours weekly. Portions of the control building, turbine building, auxiliary building and outside areas were visited. The observations included safety related tagout verifications, shift turnovers, sampling programs, housekeeping and general plant conditions. Additionally, the inspectors observed the status of fire protection equipment, the control of activities in progress, the problem identification systems, and the readiness of the onsite emergency response facilities.

The inspectors observed health physics managements involvement and awareness of significant plant activities, and observed plant radiation controls. Periodically the inspectors verified the adequacy of physical security control. Additionally, senior plant management was observed making routine tours of the plant.

The inspectors reviewed safety related tagouts 903636 (TIPS), 903805 (Startup level control valve), and 903774 (500KVA-R27J5230) to ensure that the tagouts were properly prepared, and performed. Additionally, the inspectors verified that the tagged components were in the required position.

The inspectors reviewed the activities associated with the events listed below:

On November 22, 1990, during surveillance 06-IC-1C71-R-0013, Reactor Mode Switch Interlock Function Test, a full scram and MSIV isolation occurred. The plant was in mode 4, cold shutdown at the time of the event. The surveillance had the mode switch in run and a half MSIV isolation on channel A due to a main steam line pressure low trip. The other three channels had a simulated high main steam line pressure using three transmation units (calibrated current sources). The transmation unit connected to the D channel failed, causing the scram and MSIV isolation. The battery in the transmation unit had drained to the point where the unit stopped functioning. Prior to the surveillance I&C had verified that the units had adequate battery capacity. The unit was replaced and the surveillance was completed satisfactory.

On November 24, 1990, the plant entered mode 2, startup, and was declared critical at 0833. The operators opened the inboard MSIVs when reactor pressure reached six psig. Reactor vessel level began to decrease due to the steam line drains being open. A second control rod drive pump was started to prevent the level decrease. The condensate system was not ready to supply water to the vessel due to low condensate temperature. The second CRD pump recovered reactor vessel level, but caused excessive cooling water differential pressure across the CRDMs. The excessive differential pressure caused 12 control rods to drift in 1 to 3 notches, 6 rods in the same group were out of sequence with the startup pull sheets. Technical Specification 3.1.4.2, only allows three rods out of sequence in the same group. With more than 3 rods out of sequence, additional rods motion is prohibited except by scram. The apparent cause of the manual scram was due to poor planning. The licensee appeared to rushed the startup without ensuring the plant was capable or prepared for the next evolution. The plant commenced its second reactor startup in this inspection period and achieved criticality at 1848 on November 24, 1990.

On November 24, 1990 the RWCU system isolated during the transfer from the pre-pump mode of operation to post-pump mode of operation. The system isolated on a high delta-flow isolation signal. The operator began shifting from the pre-pump mode to post-pump mode at 25 psig reactor pressure; however, the Integrated Operating Instruction (IOI) 03-1-01-1, Cold Shutdown to Generator Carrying Minimum Load, required the transfer after 100 psig. Additionally the operator started the second RWCU pump at a reactor pressue less than 100 psig. This was contrary to IOI which required placing the second pump in service at approximately 200 psig. These actions caused flow perturbations which resulted in a high differential flow signal being present after the 45 second timer had timed out. The failure to follow procedure is a violation of TS 6.8.1.a and will be documented as violation 90-25-01.

On November 27, 1990, during the performance of surveillance 06-OP-1P75-M-0001, Standby Diesel Generator 1A (Division 1) monthly functional test, large oscillations of generator field voltage, field current and output current were displayed on the local control panel with the generator loaded at 5000 KW. The generator output breaker tripped open at approximately 1255 due to the oscillations. The diesel continued to run until it was manually shutdown for corrective maintenance. Investigation into this failure revealed that the generator field voltage slip rings displayed more than expected amounts of carbon deposits. Upon cleaning the generator slip rings, a test run was performed successfully. This failure was classified as a valid failure. The frequency of the generator carbon brush and slip ring inspection was increased from an annual to a quarterly inspection.

On December 10, 1990, at approximately 1033 a reactor scram occurred on low reactor vessel water level due to an instrument air piping rupture. The loss of instrument air caused the condensate and feedwater system minimum flow valves to fail open which diverted feedwater to the main condenser. Systems responded as expected. The condensate system minimum flow valves, N19F504 A/B failing open caused a trip of the reactor feed pumps on low suction flow. High Pressure Core Spray (HPCS) auto initiated and RCIC was manually initiated. Vessel water level reached a minimum of -72 inches prior to the restart of the "B" reactor feed pump. Investigation by the licensee of the root cause of the ruptured instrument air piping revealed that an elbow joint downstream of the inlet isolation valve on the turbine building instrument air station had a failed soldered joint. The failure occurred on a copper 2 inch 90 degree elbow. The elbow connection was made using a 50-50 tin-lead solder. The elbow joint was reworked and verified acceptable by ultrasonic test (UT). Reviews by the licensee showed that 9 other connections had been made during refueling outage 4. "Snoop" and UT was completed on these solder connections. One additional connection was identified as unsatisfactory. On December 12, 1990, the unit commenced startup and achieved criticality at 2147.

On December 18, 1990, following an increase in vibration on the B recirculation pump to approximately 20 mils, a controlled reactor shutdown was initiated. A low reactor water level scram occurred at approximately 18% reactor power due to the loss of feedwater control. The plant was taken to cold shutdown to investigate the cause of the high vibration on B Recirculation pump (paragraph 6). An immediate investigation was conducted to determine the cause of the reactor feed pump A malfunction which resulted in the feed pump turbine tripping on high discharge pressure. The licensee investigation revealed that the operator at the control station noticed the A feed pump minimum flow valve, N21-F503A, cycling from full open to full closed. The decision was made to transfer to the startup level controller. The startup level control valve, N21F513 closed partially and RFP flow dropped below the minimum flow setpoint and the flow to the vessel started dropping to the low level trip setpoint. Although RCIC automatically started injecting into the vessel the reactor scrammed on low level at approximately 2116. All safety system performed as expected during the transient. The minimum water level reached was -25 inches as indicated on the wide range level instrumentation. The licensee took as found data on the A and B reactor feedwater pump High Speed Stop (HSS) and Low Speed Stop (LSS) versus speed setting piston position data. The results indicated that the HSS on the EAP actuator had been changed from 1.6 to 1.43 inches for the stroke of the speed setting piston on the B RFPT, but not on the A RFPT limit switch. The proper setting was performed on the A RFPT limit switch and a 2 inch level setpoint test was performed at approximately 17 percent power to verify stable operation.

No violations or deviations were identified.

4. Maintenance Observation (62703)

During the report period, the inspectors observed portions of the maintenance activities listed below. The observations included a review of the MWOs and other related documents for adequacy; adherence to procedure, proper tagouts, technical specifications, quality controls, and radiological controls; observation of work and/or retesting; and specified retest requirements.

MWO	DESCRIPTION
23654	Lube standby diesel lube oil heater pump.
25094	Take sample of oil from standby diesel starting air compressor.
26532	Inspection of Glyptol.
30090	Trouble shoot standby diesel generator 11.
RT9969	Inspect slip ring on generator.

No violations or deviations were identified. The observed activities were conducted in a satisfactory manner and the work was properly performed in accordance with approved maintenance work orders.

5. Surveillance Observation (61726)

The inspectors observed the performance of portions of the surveillances listed below. The observation included a review of the procedures for technical adequacy, conformance to technical specifications and LCOs; verification of test instrument calibration; observation of all or part of the actual surveillances; removal and return to service of the system or component; and review of the data for acceptability based upon the acceptance criteria.

06-IC-1B21-M-1003,	Reactor Vessel Low/High Water Level Functional Test, Channel B.
06-IC-1C11-M-0003,	Scram Discharge Volume High Water Level Float Switch (RPS) Calibration, Channel C.
06-ME-1M10-R-0003,	Drywell Bypass Leakage Rate Test.
06-OP-1E51-Q-0003,	RCIC System Quarterly Pump Operability Verification.

No violations or deviations were identified. The observed surveillance test were performed in a satisfactory manner and met the requirements of TS.

6. On-site Followup on Byron-Jackson Pump Shaft Crack at Grand Gulf (93702)

Following an unscheluded shutdown of the plant on December 18, 1990, the licensee investigated the caused of high vibration on the B recirculation pump. The investigation resulted in the testing and disassembly of the B recirculation pump. The result of the licensee investigation revealed a thru-wall crack on the B recirculation pump shaft as documented in MNCR 287-90 in the transition area. The crack extended 150 degrees around the shaft.

In May 1989, the B shaft was found with a through-wall crack (1.35") which extended approximately 320 degrees around the shaft and the A shaft had a depth crack (.950"). Details of the May 1989 shaft crack is documented in Inspection Report 89-15. The original shafts were sent to General Electric (GE) for analysis to determine the root cause of the failure. GE results indicated that through wall crack propagation resulted from cracks that were thermally initiated by a high cycle thermal fatigue mechanism in the region where seal purge flow interacts with hot reactor water. The propagation of initial cracks to shaft failure (i.e. through wall) is a result of occasional mechanical loading at pump high speed low flow (~40%) conditions. Under these conditions, GE concluded that the pump is pushing water against the partially closed flow control valve. This creates a back pressure on the pump impeller and shaft. A shaft with a crack experiences a degradation of stiffness in the same plane as the crack. This allows the shaft to bow when exposed to this mechanical stress which in turn can cause the crack to propagate to a through wall condition.

Based on assessments by GE, an independent consultant and NPE, a decision was made to defer pump shaft change outs from RF04 to RF05. The pump internals had already been purchased in anticipation of changing them out during RF04. Although the change out is considered a "like for like" replacement, the new impeller has a diameter 1/8 inches less than that of the original impellers. The radial difference is 1/16 inch for a normal impeller diameter of 34". The dimensional tolerance on the impeller radius is 1/64". The differences has been addressed by Byron Jackson and the licensee through a safety evaluation and determined to be insignificant. The mixing of hot reactor water with seal purge (cold water) in the transition region of the pump shaft was determined to be the most probable cause of the through wall cracks.

GE SIL No.- 511 recommends that reduced seal purge flow be considered for GE BWR recirculation pump manufactured by B-J to reduce the severity of thermal stresses in pump shafts and covers. GE recommended zero purge flow be used for future pump operation to reduce the severity of cyclic thermal stresses at the shaft and heat exchanger. The licensee implemented this change through revisions to procedures. A safety evaluation was performed to address the implications relative to the provisions of 10CFR50.59 for operation with zero seal purge flow to the reactor recirculation pump shaft seal assemblies.

The plant commenced startup on December 29, 1990 and was at operational condition 1, Power Operation, at 0437 on December 30, 1990.

7. Reportable Occurrences (90712, 92700)

The event reports listed below were reviewed to determine if the information provided met the NRC reporting requirements. The determination included adequacy of event description, the corrective action taken or planned, the existence of potential generic problems and the relative safety significance of each event. The inspectors used the NRC enforcement guidance to determine if the event met the criterion for licensee identified violations.

(Closed) LER 90-021, Uncontrolled Lowering of a Fuel Bundle. On October 24, 1990, during a fuel move, a fully grappled fuel bundle was lowered into the vessel in an uncontrolled manner. This was caused by independent failures of redundant refueling equipment brake systems. This event was documented in NRC inspection report 90-23. This LER is closed.

(Closed) LER 90-022, Loss of Shutdown Cooling due to Inadequate Procedure. This event was documented in NRC inspection report 90-23. This LER will be administratively closed and the corrective actions tracked under violation 90-23-02.

The licensee made a preliminary 10 CFR 50.72 report on the as found test results for the 20 MSRVs removed during refueling outage four. Fifteen of twenty MSRVs failed the technical specification test criteria of plus or minus 1 percent of the name plate set pressure. Three of the 15 valves failed at greater than plus or minus 3 percent of set pressure. All 15 valves failed low and none of the valves were reinstalled in the plant. The valves will be inspected at a later date.

8. Action on Previous Inspection Findings (92701, 92702)

(Closed) Inspector Followup Item 88-25-02, Review the corrective actions on standby liquid control system safety system functional assessment. The licensee identified 96 items for resolution during the assessment. All items have been resolved, with the exception of three, determination of heat trace operability, inadequate temperature control band for the SLCS over the full range of sodium pentaborate concentration, and the sodium pentaborate saturation temperature listed in the annual FSAR update appears to conflict with the technical specifications. The licensee addressed these items by submitting a proposed TS change in June 1989 and a revised submittal in May 1990. This submittal is under review by the NRR. The resolution of these final items has been addressed by the proposed TS amendment process. This inspector followup item is closed.

(Closed) Inspector Followup Item 89-23-02, Review corrective action on instrument air dew point. The licensee has modified the instrument air dryer muffler system. The air dryer system is maintaining dew point below the -40 degree requirement. This item is closed.

(Closed) Inspector Followup Item 90-11-01, Relabel the components in diesel generator panel P400 and P401. The components have been labeled. This item is closed.

9. Exit Interview (30703)

The inspection scope and findings were summarized on January 4, 1991, with those persons indicated in paragraph 1 above. The licensee did not identify as proprietary any of the materials provided to or reviewed by the inspectors during this inspection. The licensee had no comment on the following inspection findings:

Item Number	Description and Reference
VIO 90-25-01	Failure to follow procedure for placing in service the second RWCU pump, paragraph 6.

10. Acronyms and Initialisms

ADHRS- Alternate Decay Heat Removal System
ADS - Automatic Depressurization System
APRM - Average Power Range Monitor
ATWS - Anticipated Transient Without Scram
BWR - Boiling Water Reactor
CRD - Control Rod Drive
CRDM - Control Rod Drive Mechanism
DCP - Design Change Package
DG - Diesel Generator
ECCS - Emergency Core Cooling System
ESF - Engineering Safety Feature
FCV - Flow Control Valve
FSAR - Final Safety Analysis Report
HPCS - High Pressure Core Spray
HPU - Hydraulic Power Unit
I&C - Instrumentation and Control
IFI - Inspector Followup Item
IOI - Integrated Operating Instruction
LCO - Limiting Condition for Operation
LER - Licensee Event Report

LPCI - Low Pressure Core Injection
LPCS - Low Pressure Core Spray
MSIV - Main Steam Isolation Valve
MWO - Maintenance Work Order
NPE - Nuclear Plant Engineering
NRC - Nuclear Regulatory Commission
PDS - Pressure Differential Switch
P&ID - Piping and Instrument Diagram
PSW - Plant Service Water
QDR - Quality Deficiency Report
RCIC - Reactor Core Isolation Cooling
RHR - Residual Heat Removal
RPS - Reactor Protection System
RWCU - Reactor Water Cleanup
RWP - Radiation Work Permit
SLCS - Standby Liquid Control System
SOI - System Operating Instruction
SRV - Safety Relief Valve
SSW - Standby Service Water
TCN - Temporary Change Notice
TIPS - Traversing In-Core Probe System
TS - Technical Specification