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

Report No.: 50-416/89-16

Licensee: System Energy Resources, Inc.
Jackson, MS 39205

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

Facility Name: Grand Gulf Nuclear Station

Inspection Conducted: May 20 - June 16, 1989

Inspectors: _____ 6/27/89
H. O. Christensen, Senior Resident Inspector Date Signed

_____ 6/27/89
J. L. Mathis, Resident Inspector Date Signed

Approved by: _____ 6/27/89
F. S. Cantrell, Section Chief Date Signed
Division of of Reactor Projects

SUMMARY

Scope:

The resident inspectors conducted a routine inspection in the following areas: operational safety verification, maintenance observation, surveillance observation, engineering safety features (ESF) system walkdown, 10 CFR Part 21 procedures, action on previous inspection findings, and reportable occurrences. The inspectors conducted backshift inspections on May 20, 21, 23, 24, 29 and June 13, 1939.

Results:

Within the areas inspected two violations were identified: Failure to take adequate corrective action to prevent RCIC system isolations, paragraph 9, and failure to maintain post-accident sample system design control, adequate training, and perform required samples, paragraph 9.

The post accident sample system problems appear to be related to the licensee treating the system as a non-safety, non-technical specification system (no limit conditions for operations). Controls placed on safety-related systems are more inclusive and the plant staff appears to focus more attention to the safety related system status as compared to non-safety related system.

The recirculation pump shaft replacement outage was well managed.

REPORT DETAILS

1. Persons Contacted

Licensee Employees

*J. G. Cesare, Director, Nuclear Licensing
W. T. Cottle, Vice President of Nuclear Operations
D. G. Cupstid, Superintendent, Technical Support
*L. F. Daughtery, Compliance Supervisor
*J. P. Dimmette, Manager, Plant Maintenance
S. M. Feith, Director, Quality Programs
*C. R. Hutchinson, GGNS General Manager
F. K. Mangan, Director, Plant Projects and Support
R. H. McAnulty, Electrical Superintendent
A. S. McCurdy, Technical Asst., Plant Operations Manager
*L. B. Moulder, Operations Superintendent
J. H. Mueller, Mechanical Superintendent
S. F. Tanner, Manager, Quality Services
L. G. Temple, I & C Superintendent
F. W. Titus, Director, Nuclear Plant Engineering
*M. J. Wright, Manager, Plant Support
J. W. Yelverton, Manager, Plant Operations
G. Zinke, Superintendent, Plant Licensing

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

*Attended exit interview

F. S. Cantrell, Section Chief, Division of Reactor Projects, was on site May 30 and 31, 1989, to conduct a plant tour and hold discussions with the resident inspectors.

2. Plant Status

Unit 1 began the inspection period in a recirculation pump maintenance outage. On May 31, 1989, Unit 1 restarted and returned to power operations.

3. Operational Safety, (71707)

The inspectors were cognizant 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, status of operating systems, tags and clearances on equipment controls and switches, annunciator alarms, adherence to limiting conditions for operation, temporary alterations in effect, daily journals and data sheet entries, control room manning, and access controls. This

inspection activity included numerous informal discussions with operators and their supervisors.

On a weekly bases selected engineered safety feature systems were confirmed operable. The confirmation was made by verifying that accessible valve flow path alignment was correct, power supply breaker and fuse status was correct, and instrumentation was operational. The following systems were verified operable: Suppression pool makeup, control rod drive system, HPCS and LPCS.

General plant tours were conducted on a weekly basis. 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, the status of fire protection equipment, control of activities in progress, problem identification systems, and the readiness of the onsite emergency response facilities.

The inspectors observed health physics management involvement and awareness of significant plant activities, and observed plant radiation controls. Additionally the inspectors verified the adequacy of physical security controls.

The inspector reviewed safety related tagout 892996 (Equipment drain sump pump). The review ensured that the tagout was properly prepared, and performed. Additionally, the inspectors verified that the tagged components were in the required position.

The inspectors verified that the following containment isolation valves were in there correct lineup; E22-F035, E22-F022 and E12-F339.

The inspectors noted that senior plant management makes routine tours to the plant and the control room.

The inspectors reviewed activities associated with the failure of the B recirculation pump shaft. On May 15, 1989, the plant was taken to cold shutdown due to high vibration on the B Recirculation Pump. Upon disassembly, a 300 degree crack was discovered on the lower pump shaft. Both A and B recirculating pump rotating elements were replaced. The details of the shaft failure are documented in NRC inspection report 50-416/89-15. The recirculation pump maintenance outage was well planned, scheduled and managed. The plant conducted an orderly restart on May 31, 1989.

No violations or deviations were identified.

3. Maintenance Observation (62703)

During the report period, the inspectors observed portions of the maintenance activities listed below. The observations included a review of the MWO 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
DCP 86/0085	Extension of the upper containment pool weir wall
M 93729	SSW B basin fan D repair
M 93647	Recirculation pump internal inspection
M 93480	Recirculation pump internal removal and replacement
M 93482	Recirculation pump assemble of spare rotating assembly
EL 2128	Megger Motor (SBLC) from the breaker
EL 2110	Inspect MOV C41F001B
ME 3402	Unit 1 Instrument Air Dryer Desiccant Change Out

No violations or deviations were identified.

5. Surveillance Observation (61726)

The inspectors observed the performance of portions of the surveillances listed below. The observation included a review of the procedure 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-OP-1P75-M-0001, Standby Diesel Generator (SDG) II Functional Test, Attachment II

06-EL-1E31-M-0001, RCIC Main Steam Tunnel Isolation Delay Timer Channel A Functional Test and Calibration, Attachment 1

06-IC-1E31-M-0022, Drywell Air Cooler Condensate Flow Rate Monitoring Functional Test

06-IC-1E32-M-1002, MSIV Leakage Control System Functional Flow Test, Attachment 1

06-IC-1C34-M-0001, Reactor Vessel Water Level (Level B) MT/RFPT Trip Function Test

06-IC-1D17-A-0012, Fuel Handling Area Ventilation Radiation Monitor Calibration

06-IC-1E31-M-2003, Main Steam Line "C" High Flow Functional Test

No violations or deviations were identified.

6. Engineered Safety Features System Walkdown (71710)

The inspectors conducted a complete walkdown on the accessible portions of the standby gas treatment system. The walkdown consisted of the following: confirm that the system lineup procedure matches the plant drawing and the as-built configuration; identify equipment condition and items that might degrade plant performance; verify that valves in the flow path are in correct positions as required by procedure and that local and remote position indications are functional; verify the proper breaker position at local electrical boards and indications on control boards; and verify that instrument calibration dates are current.

The inspectors walked down the system using system operating instruction 04-1-01-T48-1, Revision 19, SBGT and P&ID M-1102 A and B, SBGT system.

The monthly operability test for SBGT systems A and B were performed satisfactory and the 18 month system logic and vacuum test was successfully performed. Additionally, the 18 month calibrations for drywell high pressure and reactor vessel water level were performed. The annual fuel handling area ventilation exhaust radiation monitor calibrations were also performed successfully.

The SBGT electrical lineup was verified by using attachment III to the system operating instruction. All electrical breakers were in the required position. The instructions component description differed from the breakers label name for all breakers. A labeling program has been implemented to correct all labeling deficiencies.

All annuciators and valve positions were in accordance with the system operating instruction.

No violations or deviations were identified.

7. 10 CFR Part 21 Inspection (36100)

The inspectors verified that procedures and controls were established and implemented for 10 CFR Part 21 requirements. The initiating document for Part 21 is through a Material Nonconformance Report (MNCR). MNCRs are used to document discrepancies concerning material-related documentation, i.e., test results, certification, and etc. Administrative Procedure 01-S-03-3, Material Nonconformance Reports designates Nuclear Plant Engineering (NPE) as the organization responsible for evaluating whether deficiency or nonconformances constitute reportability pursuant to 10 CFR Part 21. Additionally, Quality Programs screens all nonconformance reports for potential Part 21. Quality Assurance Procedure (QAP) 6.40, Potential Reportable Deficiency Screening, provides guidance to be used by the screening reviewer for Part 21.

The inspectors reviewed procedure 01-S-09-1, Revision 24, Procurement of Materials and Services, which requires 10 CFR 21 be required on all Quality Level 1 and 2 material item procurements and on all Quality Level 1, 2 and 3 service contracts. The inspector performed a random sample of

purchase orders written after January 6, 1987. All four purchase order contained the 10 CFR 21 applicability statement. In addition the inspectors selected two evaluated deviations or nonconformances not resulting in a report to the commission to verify the following:

- The item was identified for evaluation consistent with established procedures.

- The information and data used in the evaluation appear to be factual and complete.

- The nonconformance was evaluated, or forwarded to the purchaser for evaluation consistent with established procedures.

Overall it appears that the licensee has an effective program in place for evaluating 10 CFR 21 requirements.

8. 10 CFR 50.59 Safety Evaluation

In a May 22, 1989, NRC letter to SERI, NRR documented the NRC staff's safety evaluation results for five SERI 10 CFR 50.59 safety evaluations. Corrective actions were recommended in three areas:

- Revision of the SERI safety evaluations NPE-86-279 and PLS-86-123 to include adequate bases to support a determination that the changes do not involve unreviewed safety questions.

- Revision of the UFSAR to include information to show how safety significant cranes meet NUREG-0612, as discussed in NRC evaluation of NPE-86-279.

- Revision of the surveillance procedure for TS 4.6.6.1.b to require that drawdown test of secondary containment be run with the primary containment hatch open as discussed in the NRC evaluation of PLS-86-136.

The resolution of the above recommendation will be an inspector followup item 89-16-01.

9. Reportable Occurrences (90712 & 92700)

The below listed event reports were reviewed to determine if the information provided met the NRC reporting requirements. The determination included adequacy of event description and corrective action taken or planned, existence of potential generic problems and the relative safety significance of each event. Additional inplant reviews and discussions with plant personnel as appropriate were conducted for the reports indicated by an asterisk. The event reports were reviewed using the guidance of the general policy and procedure for NRC enforcement actions, regarding licensee identified violations.

a. (Closed) LER 89-005, Reactor Core Isolation Cooling System Isolations on Indicated High Steam Line Flow. The RCIC isolations of April 29, and May 8, 1989, were documented in NRC inspection report 89-14. Upon completion of the investigation, it was determined that the isolations were caused by a spurious high flow signal produced by pressure oscillations in a sensing line for a RCIC steam line differential pressure transmitter. The sensing lines were backfilled with demineralized water to reduce the amount and amplitude of the oscillation and the damping pots were increased to give a stable signal to the transmitters. Similar isolations occurred in December 1984, as reported under LERs 84-56 and 84-57. The 1984 correction was to adjust the damping of the transmitters to depress the level of oscillations. The adjustments were completed by early 1985. During refueling outage three, I&C technicians replaced the electronic circuit boards for the transmitters. The work package required the damping pots to be set to the as found setting. The old pots were found to be set at minimum damping, therefore the technicians set the new circuit board pots to minimum. A records search of work performed on the transmitters did not reveal when or how the old pots were readjusted. The failure to adequately control the damping values for the RCIC transmitters is a violation of 10 CFR 50, Appendix B, Criterion XVI Corrective Actions. The LER 89-005 will be administratively closed and corrective actions will be tracked under violation 89-16-02.

b. On June 7, 1989, the shift superintendent made a one hour report per 10 CFR 50.72.(b).(v) on the loss of post-accident sampling assessment capability. While performing a post modification retest, for DCP 87-4018, PAS system atmospheric and liquid panel modification, it was determined that two valves, P33F719 and P33F720, which were required to be deleted by the design change were installed and isolated, preventing the capability to take a post-accident liquid radionuclide sample. Further investigation determined other design package deficiencies, which include: Inadequate post modification system walkdowns, several valves and piping not in accordance with as-built drawing, inadequate post modification testing, performed system retest approximately one month after completion of modification and at 100% reactor power; and inadequate design package review and implementation. The package contained vendor equipment numbers which were not converted to SERI system numbers and the package was not translated into adequate work instructions, that is valves P33F719 and P33F720 were left installed contrary to design intent.

It was also determined that the system had not previously been operated in accordance with the plants license condition, TS, or emergency plan. Licensing Condition 2.C.(33).(c), requires SERI to incorporate the requirements of Safety Evaluation Report, Supplement 4 (SSER 4), into procedures. SSER 4, Section 22.2,II.B.3, requires a post-accident sampling program be performed on a semiannual basis and consists of obtaining and analyzing reactor coolant, suppression pool, and RHR samples chemically and radiochemically by persons

responsible for post-accident procedures. SERIs Emergency Plan, Section 7.6.4, states that systems are installed to obtain samples from the following locations: "...RHR A and B; drywell atmosphere; and suppression pool." TS 6.8.3.c, requires a post-accident sampling program be established, implemented and maintained which will ensure the capability to obtain and analyze a sample under accident conditions, training of personnel, procedures for sampling and analysis,and provisions for maintenance of the equipment.

A review of past sampling data, January 1986 to June 1989, indicated that semiannual samples were not conducted on RHR or suppression pool and the RHR B and drywell sample paths were never tested. Additionally, discussions with the system engineer and plant chemistry indicated that they believed a sample could only be taken during full system pressure. Chemistry personnel attempted to take samples on April 24 and 27,and May 17, 18, and 19, 1989, when the plant was shut down and depressurized. Only small amounts of water were obtained, this led them to believe that a sample path was available. The system engineer and plant chemist decided to sample at pressurized conditions due to the sample difficulties at depressurized conditions. The PAS system was declared operable on April 27, 1989. The plant operated at various power levels for approximately 20 days prior to conducting a post modification sample test on June 6, 1989. Chemistry was unable to obtain a sample and on June 7,1989, the valves P33F719 and P33F720 were discovered in the system. The licensee performed a dose rate assessment calculation to determine the radiation level that would be present during accident conditions at the area of the two isolated valves. The calculation determined an exposure field of 800 rem per hour on contact. (Other locations are available that could be used to obtain a sample although the exposure maybe higher than would be expected from an operable PAS system.)

The failure to maintain the PAS system operable, to conduct semiannual samples of the RHR and suppression pool, to demonstrate the drywell and RHR B sample paths, and to adequately train the system engineer and plant chemistry in PAS system operations is a violation of TS 6.8.3.c, and 10 CFR 50, Appendix B, Criterion III. This will be documented as violation 89-16-03.

c. On June 7, and 8, 1989 during a hotline phone check, none of the offsite emergency agencies could be contacted by the TSC or the control room emergency notification system. The licensee immediately notified the telephone company to correct the problem. This incident was also called into the NRC as a one hour reportable event per 10 CFR 50.72(b)(1)(v).

d. On June 8, 1989 a power outage in Claiborne County led to 22 sirens being without power, the licensee reported the incident to NRC per 50.72(b)(1)(v).

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

(Closed) IFI 86-02-03, Inspector Followup Item, NRR conducted review of BWR technical specifications (TS) governing requirements for safety relief valve (SRV) operability and if changes to TS should be made. NRR concluded that they do not have a safety concern for those BWR reactors which have TS allowing unlimited operation with more than one SRV out of service because: (1) the number of SRVs required to be operable by TS are based on the overpressure protection analysis as described in the Standard Review Plan 5.2.2, (2) the number of SRVs out of service up to the maximum allowed by TS is unlikely to affect fuel cladding integrity during transients and accidents described in the FSAR Chapter 15, and (3) there are no regulatory requirements for an ATWS analysis. However, the discrepancy between the TS limiting conditions for operation with SRVs out of service and the GE recommended administrative procedure should be resolved.

As a followup to this evaluation, NRR will address the inconsistency between the TS and the GE recommended procedure with the BWR Owners Group as a part of the TS Improvement Program.

The Generic Communications Branch, NRR has determined that no action is warranted to inform affected licensees because operation with the present TS is not a safety significant concern. This item is closed.

(Closed) IFI 89-04-04, Incorporate GE SIL 319 recommendations into PM program. Maintenance Planning & Scheduling System initiated task card ME6044 to inspect RCIC Turbine Drive Gear Assembly every 18 months. Procedure 07-2-14-301, RCIC Turbine Drive Gear Assembly Inspection, is the procedure used by mechanical maintenance personnel to inspect the RCIC gear assembly. This item is closed.

(Closed) IFI 85-09-05, Design change on precoat filter isolation causing scram. The inspectors reviewed DCP 84/3000 and MWO 52146 which provided an automatic start of the instrument air compressor on low air receiver pressure and allowed operators to restart the air compressor remotely from the control room whether it is being operated in the automatic or manual mode. This item is closed.

(Closed) URI 86-01-02, Review test exceptions to level 1 startup test acceptance for 10 CFR 50.59 consideration. The inspectors reviewed IPC 86/0296 and IPC 86/363, which reviewed all startup exceptions written to document level 1 criteria failures. The findings of the review verified that no further FSAR changes were required. This item is closed.

(Closed) Violation 88-07-03, Violation, Failure to follow procedure for procurement of materials. The licensee revised the warehouse inventory computer system to reflect the voltage level and type as part of the model number for all ASCO solenoid valves in stock. Future ASCO solenoid valves will be stored according to voltage level and type. This item is closed.

(Closed) IFI 88-19-02, Inspector Followup Item, Implementation and repair of 3/4" HCD-73 line. MWO M91252 implemented MCP 89/1011 during RF03 to repair the vent line for valve Q1B33F023A and repair the 4" drain hub. This MCP modified the vent line and replaced valves B33F025A and B33F026A with identical valves. This item is closed.

(Closed) IFI 87-16-05, Installation and testing of warning system for high noise areas of the plant. Visual alarms were installed and tested under DCP 84/0231. This item is closed.

11. Exit Interview (30703)

The inspection scope and findings were summarized on June 16, 1989, 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 General Manager stated that they view the PAS system problem very serious and that it is a result of multipule breakdowns. The licensee had no additional comments on the following inspection findings:

Item Number	Description and Reference
89-16-01, IFI	Resolution of 10 CFR 50.59 reviews, paragraph 8.
89-16-02, VIO	Failure to adequately control the damping values for RCIC transmitters, paragraph 9.
89-16-03 VIO	Failure to maintain the PAS system design control, to conduct adequate training and perform required samples, paragraph 9.

12. Acronyms and Initialisms

ADHRS- Alternate Decay Heat Removal System
ADS - Automatic Depressurization System
APRM - Average Power Range Monitor
CRD - Control Rod Drive
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
LCO - Limiting Condition for Operation
LER - Licensee Event Report

LPCI - Low Pressure Core Injection
LPCS - Low Pressure Core Spray
MNCR - Material Nonconformance Report
MWO - Maintenance Work Order
NPE - Nuclear Plant Engineering
NRC - Nuclear Regulatory Commission
PASS - Post Accident Sample System
PDS - Pressure Differential Switch
P&ID - Piping and Instrument Diagram
PSW - Plant Service Water
QAP - Quality Assurance Procedure
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
SBLC - Standby Liquid Control
SDG - Standby Diesel Generator
SERI - System Energy Resource Incorporation
SOI - System Operating Instruction
SSW - Standby Service Water
TCN - Temporary Change Notice
TS - Technical Specification
TSC - Technical Support Center
UFSAR- Updated Final Safety Analysis Report