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

Report Nos.: 50-321/90-22 and 50-366/90-22

Licensee: Georgia Power Company
P. O. Box 1295
Birmingham, AL 35201

Docket Nos.: 50-321 and 50-366 License Nos.: DPR-57 and NPF-5

Facility Name: Hatch 1 and 2

Inspection Conducted: October 15-19, 1990

Inspectors: C. Smith — 11-19-90 Date Signed

M. Thomas — 11-19-90 Date Signed

Accompanying Personnel: F. Jape on October 17-19, 1990

Approved by: for F. Jape, Section Chief — 11/19/90 Date Signed

SUMMARY

Scope:

This routine announced inspection was conducted in the areas of Design Changes and Modifications and followup on SSFI open items.

Results:

In the areas inspected, violations or deviations were not identified.

Review of completed and partially completed design change packages revealed that plant modifications are made in a controlled and technically adequate manner. Licensee management's involvement in assuring quality is demonstrated by the development and implementation of an Engineering Quality Improvement Program intended to improve the quality of engineering deliverables. Review of selected portions of this program verified that the program has been effective in meeting specified objectives. Additionally, a Design Change Request Closeout Program has been effective in reducing the number of plant modification packages which were still open.

REPORT DETAILS

1. Persons Contacted

Licensee Employees

*K. Breitenback, Engineering Supervisor
*D. Edge, Manager, Nuclear Security
*D. Davis, Manager, Plant Administration
*P. Fornel, Manager, Maintenance
*O. Fraser, Supervisor, SAER
*M. Gauge, Manager, Outage and Planning
G. Goode, Manager, Engineering Support
*J. Hammonds, Supervisor, Regulatory Compliance
*W. Kirkley, Supervisor, HP/Chemical Engineering
*J. Lewis, Operations Manager
*D. Madison, Engineering Manager, Hatch Corp.
*T. Moore, Assistant General Manager, Plant Support
*D. Reid, Acting General Manager
*J. Robertson, Jr., Acting Engineering Support Manager
*S. Tipps, Manager, Nuclear Safety Compliance
*R. Zavadoski, Manager, HP/Chemistry

Other licensee employees contacted during this inspection included engineers and administrative personnel.

Other Organizations

B. Garner, Manager, Hatch Project, SCS
*G. McGaha, Design Manager Hatch, SCS
*K. Khianey, Project Engineer, Bechtel

NRC Resident Inspectors

*R. Musser, Resident Inspector

*Attended exit interview

Acronyms and initialisms used throughout this report are listed in the last paragraph.

2. Design Changes and Modifications (37700)

a. DCR 1H89-192, RCIC Low Speed Bypass Line

This DCR was implemented to provide an orificed bypass test line with a motor operated valve around the steam admission valve in order to reduce the severity of the RCIC start transient, thereby reducing the possibility of an overspeed trip. This design change was divided into two parts. The first part was completed under DCR 1H89-192 which involved the mechanical portions and the routing of electrical cables. The second part will be completed under DCR 1H90-072 which involves completing the electrical logic changes and cable terminations to make the system operable. DCR 1H90-072 had not been completed at the time of this inspection.

The inspector performed a field walkdown to verify that DCR 1H89-192 was installed in accordance with applicable design documents.

b. DCR 1H89-249, Diesel Generator Check Valves

The modification involved replacing the emergency diesel generator (EDG) starting air compressor discharge check valves and the check valves installed in each supply header to the air receivers. The old check valves were replaced with ones that are believed to provide greater assurance of being leak tight. This modification also involved installing a test connection to allow venting of the header during check valve testing.

The inpsector performed field walkdowns of EDGs 1A, 1B, and 1C to verify that the check valves were installed in accordance with design documents. This modification is scheduled to be implemented for Unit 2 during the next Unit 2 refueling outage.

While reviewing DCR 1H89-249, the inspector observed that the acceptance criteria specified in the post modification test special purpose procedure 34SP-111689-DC-1-OS, DG Air Start Check Valve Functional Test, was greater than the total four hour leakage specified by design engineering. Design engineering specified that the maximum leakage for four hours should be less than 75 psi while the special purpose procedure acceptance criteria specified 95 psi. The inspector discussed this issue with licensee personnel who stated that the 95 psi value stated in the special purpose procedure was used in error. The licensee initiated procedure revisions to correct the special purpose procedure prior to implementation for Unit 2, and to correct surveillance procedure 34SV-R43-014-OS, Diesel Generator Air Start Check Valve Functional Test.

The inspector reviewed the completed test results for EDGs 1A, 1B, and 1C and found that the maximum four hour leakage for any of the check valves tested was 16 psi, which was well below the maximum leakage specified by either design engineering or the value specified in the special purpose procedure. This demonstrated that the check valves were operating properly.

c. DCR 1H90-079, LPCI Injection Valves

This modification involved replacing the valve stem and valve disc for RHR LPCI injection valves 1E11-F017A and 1E11-F017B due to damage to the valve stem and valve disc of both valves. The exact replacement materials of the original stem and disc were unavailable, so this change was processed by design engineering who evaluated the adequacy and acceptability of the replacement material.

d. DCR 1H90-153, Diesel Generator Undervoltage Relay Time Dial Setting

This modification involved changing the time dial setting for undervoltage relays CV-7 for each Unit 1 EDG in order for the EDG to be able to energize their emergency busses in less than or equal to 12 seconds on an automatic start signal. This issue was discovered by site emergency personnel during efforts to develop the surveillance test procedure to meet this new proposed TS requirement.

The inspector discussed this item with licensee personnel who stated that this issue has been discussed in detail with NRC Regional and Headquarters management. The actions taken to resolve this issue are discussed in greater detail in Licensee Event Report (LER) 50-321/1990-017 and NRC Inspection Report 50-321, 366/90-20.

e. DCR 2H90-154, Diesel Generator Undervoltage Relay Time Dial Setting

This modification involved changing the time dial setting for the undervoltage relays for the Unit 2 EDGs as discussed above under DCR 2H90-153.

f. DCR No. 88-294, 600 Volt Switchgear

RER No. 87-514 was written on September 25, 1987, to request evaluating the installation of micro-verse trip elements to replace existing EC type trip elements. An attachment to the RER documented severe problems with AK type breakers equipped with EC trip devices and provided a cost-benefit analysis which favored use of the micro-verse trip devices. The final disposition of RER resulted in the preparation of DCR No. 88-294 with stated design objectives of (1) improving coordination between circuit breakers installed in GE 600V switchgears 1R23S003 and 1R23S004 and downstream loads, and (2) reducing maintenance cost associated with the present trip device.

The inspector reviewed the DCP and verified that a 10 CFR 50.59 safety evaluation screening had been performed with no USQ identified. Design basis calculations numbers SNC-85-098, Seismic Evaluation of Micro Versa Trip, Revision 3, and SEN 89-010, Micro Versa Trip Coordination Study, Revision 2, was verified as having been completed. Selected portions of calculation SEN 89-010 were reviewed and verified to be technically adequate. Additionally, post-modification test requirements specified in surveillance procedure 52SP-111688-RW-1-05 were verified as having been completed.

Refurbishment of the circuit breakers and installation of the micro-versa trip units was performed off-site by the OEM. The work was accomplished under PO No. P-02757 which was reviewed by the inspector to verify that a applicable technical and a quality requirements had been imposed on the vendor. A field walkdown of the installed circuit breakers was performed by the inspector to verify agreement between plant switchgear lineup and design drawings, and calculated trip setpoints versus actual setpoints of selected circuit breakers.

g. DCR No 88-349T, 100 Percent Stator Ground Fault Protection Relay

DCR No. 88-349T was written to remove the 100 percent stator ground fault relay, 64 GSX, from service to facilitate troubleshooting the relay. This temporary plant modification is non-nuclear safety related and was administered under the licensee's temporary modification program. The inspector verified by review of objective evidence that (1) disabling of the relay and removal of the NW-511-2 module was performed in a controlled manner via MWO No. 1-88-7849, (2) the TM while still installed was periodically reviewed in accordance with the TM program controls, and (3) reinstallation of the NW-511-2 module and enabling of the relay after repairs by the vendor was adequately controlled via MWO 1-89-4815. Successful completion of post installation test was verified by review of document number 57IT-S32-002-1N, Brown Boveri Type GIX 103 Relay 100 Percent Stator Ground Fault Protection, Revision 0.

h. DCR No. 82-173, RHR Time Delay Relays

SIL No. 230, Revision 2, dated July 1981, provided information concerning the time delay of GE CR 2820 time delay relays used in the core spray, RHR, and automatic depressurization system. The SIL stated that the time delay of the CR 2820 relay tends to increase after long periods in the de-energized state and recommended their replacement with Agastat time delay relays. The above DCP was prepared in response to DCR 82-173 dated May 25, 1990, and had a stated design objective of providing the RHR system logic with time delay relays that will maintain their catalog specified accuracy. The scope of the hardware changes involved the changeout of 11 time delay relays.

The inspector reviewed the DCP and determined that partial implementation of the plant modification had been completed with changeout of relays 1E11-K93A and B. The replacement relays were Struthers Dunn, model 236ABXP. A documented basis for use of the Struthers Dunn relays was not part of the DCP. Licensee management stated that, based on the poor experience with the Agastat time delay relays, a decision was made to use Struthers Dunn relays. The replacement relays are used in the logic for the RHR Heat Exchanger Bypass Valves E11-F048A and B. The inspector verified that the 10 CFR 50.59 safety evaluation bounded the scope of the design activities, that the drawings accurately reflected the hardware changes, and the elementary diagrams correctly implemented the required logic with specified time delay. Post modification testing was verified to have been successfully completed in accordance with special purpose procedure 42SP-052990-PW-1-1S, 1-K93A and B, Functional Test for DCR 82-173. One minor weakness related to the absence of installation instructions for the replacement relays was identified.

This information was added by FCR 82-173-1. A walkdown of the installed relays verified that installation was completed in accordance with design documents.

No violations or deviations were identified in the areas inspected.

3. Engineering Support Activities

a. Engineering Quality Improvement Program

Licensee management is presently implementing an Engineering Quality Improvement Program intended to improve the quality of engineering deliverables. The following selected areas were reviewed to ascertain the program effectiveness.

(1) FCR Tracking

FCRs are the administrative process used for making onsite changes to DCPs prepared by offsite engineering organizations. The licensee has developed and implemented a program to evaluate and trend FCRs generated (1) during DCP installation and (2) identified during plant/equipment walkdown. A uniform list of reason codes has been prepared for use by SCS and Bechtel pursuant to the analysis of field change data transmitted to the engineering organizations. The inspector determined that existing work practices related to FCR tracking needs has not been incorporated in SCS procedures.

Licensee management has established a goal of not more than two FCRs per DCP with assignable cause to design-engineering activities. Quarterly reports with data presented in bar chart form is prepared along with FCR tracking forms delineating corrective actions for each deficiency. At the time of the inspection, a summary report documenting the analysis of data gathered up to September 1, 1990, had been prepared for management review.

An evaluation of the effectiveness of the FCR tracking program could not be made given the relatively short time it has been in effect. Baseline performance data is presently being gathered and analysed. The effectiveness of the program will be demonstrated after it has been in effect for some reasonable time.

(2) ABN Program

ABNs are the administrative process used to document discrepancies identified between approved design drawings/documents and as-found plant installation. If required by plant administrative procedures, a deficiency card is prepared to document the discrepancy. Alternatively, an explanation will be provided on the ABN form as to why a deficiency card is not required.

The inspector determined that Hatch design configuration project had a dedicated staffing level of 61 equivalent people during 1989 and 45 equivalent people for 1990. The processing of ABNs were performed in

accordance with approved procedures and the results of the drawing update activities are as follows:

- End of 1988

  82 percent of total drawings had no outstanding ABNs

- End of 1989

  94 percent of total drawings had no outstanding ABNs

- Current 1990

  99 percent of total drawings had no outstanding ABNs.

Additional statistics presented by the licensee demonstrated that the program has been extremely effective in ensuring plant configuration. All category 1 drawings, which require 30 days to be updated, are current. There has been a significant reduction of Category 3 drawings. These drawings are updated 90 days after receipt of the third ABN or upon request, and was the largest population that contributed to lack of configuration control. The results of the drawing update are as follows:

1989 Category 3:	79,276
1990 Category 3:	457

Updated Category 3 drawings are now controlled as Category 2 drawings which ensure a more timely revision than that for Category 3. The inspector concluded that the ABN program has been effective in maintaining consistency between as design drawings and actual plant installation.

b. DCR Closeout Program

A formalized DCR closeout program was initiated in January 1990, to reduce a backlog of 186 open DCRs. Phase 1 of the program consisted of (1) defining the DCR closeout program scope, (2) review of implemented but not closed out DCRs for completeness, (3) verifying accuracy of implementation by walkdown, (4) taking corrective actions as discrepancies are identified, (5) updating plant documents, and (6) closeout of the DCRs. The program was implemented in accordance with administrative controls delineated in the following procedures/department instructions:

- Special Purpose Procedure No. 42SP-112989-OB-1-OS, DCR Processing/Voiding Detailed Instructions, Revision 0.

- Departmental Instruction DI-ENG-42-1189N, DCR Closeout/Voiding Guidelines, Revision 0.

- Procedure 42EN-ENG-001-OS, DCR Processing, Revision 7.

The inspector verified by review of objective evidence that, at the time of the inspection, the DCR closeout program had accomplished the following:

- 122 DCRs had been closed out.
- 35 DCRs had been voided in accordance with written criteria.
- five DCRs had been transferred out of the scope of the closeout program for various reasons.
- 24 DCRs were still outstanding awaiting closures.

Substantial resources were expended by licensee management to accomplish the above tasks via a dedicated group assigned to the DCR Closeout Group. Phase 2 of the DCR Closeout Program is presently in the planning stage and is being developed to process and close out partially implemented DCRs. Phase 2 will include DCRs that are outside the scope of Phase 1, and is intended to assess and complete all required actions for implementing and closeout of approximately 50 partially implemented DCRs having various classifications.

The inspectors concluded that the DCR Closeout Program has been effective in reducing the number of open DCRs identified in Phase 1.

c. Deficiency Cards and Significant Occurrence Reports

Deficiency cards are used to identify and document problems found in the plant. The DCs are initially reviewed for significance and potential impact on plant operations by the STA and applicable unit shift supervisor. Items which are determined to be significant are given a significant occurrence report number by the Nuclear Safety and Compliance Department and are assigned to various plant departments for followup and resolution. The percentage of DCs written thus far in 1990 which have resulted in SORs is approximately three percent. The licensee trends SORs and DCs on a quarterly basis.

The inspector reviewed selected SORs which have been assigned to the site Engineering Support Department. Nearly 40 percent of the 1990 SORs were assigned to Engineering Support for action. Engineering Support management stated that meeting the SOR response due dates is continually emphasized to engineering personnel. The inspector noted that the response dates for SORs assigned to engineering were met in nearly all instances. The inspector noted a few instances where the SOR responses prepared by engineering were rejected by the PRB and sent back to engineering for a variety of reasons. Subsequent responses for the applicable SORs were resubmitted and accepted by the PRB. The inspector noted that, while a few of the engineering responses to SORs were considered inadequate by the PRB, the subsequent review and approval demonstrated that the licensee's overall review process for SORs (including the PRB) was effective in ensuring that adequate resolutions for SORs were being provided by the various departments.

d. Temporary Modifications

The inspector reviewed the licensee's temporary modification program and the efforts to reduce the TM backlog. A team consisting of engineering, maintenance, and operations personnel was formed in mid-1989 to evaluate TMs that were greater than 90 days old. The team recommended removal of those TMs that could be removed, implementation of DCRs for some, and incorporation of repetitive TMs into procedures. This process resulted in the number of TMs greater than 90 days old being reduced from approximately 55 in June 1989 to 18 in September 1989. At the beginning of 1990, ten TMs greater than 90 days old were open, nine of which were related to the Unit 1 outage. As of October 1990, 15 of the 33 open TMs were greater than 90 days old. Licensee personnel stated that the main reason for reducing the number of TMs greater than 90 days is to reduce the paperwork load on the unit shift supervisors. The goal is to reduce the number of non-outage TMs to zero. The status of TMs greater than 90 days old are discussed monthly with the Plant General Manager.

The inspector reviewed selected open TMs and the licensee's administrative controls covering TMs. All safety-related TMs are required to be reviewed by the PRB prior to implementation. TMs can be installed for up to one year. All TMs that remain installed for greater than 90 days must be reviewed by Engineering Support. The unit shift supervisors review the TM logs monthly and identify to Engineering Support those TMs which require review because the TMs are greater than 90 days old. The inspector review ' all the open safety-related TMs that were greater than 90 days old a verified that both the unit shift supervisors and Engineering Support reviews were performed in accordance with the applicable administrative controls. The following safety-related TMs were reviewed:

1-89-075	Temporary Clamp to Block Off Air Leak on Transfer Canal Seal Assembly.
2-90-005	Refueling Bellows Leak Detection Alarm Removed.
2-90-006	Annunciator for 2E41-F006 Overload Removed.
2-90-015	Points 14, 16, and 20 Causing Annunciation for Recirculation Temperature Recorder.
2-90-019	Clamp Placed on Three Pin Hold Leaks of PSW Discharge Header Minimum Flow.

The inspector determined from reviewing the licensee's TM program that the licensee's efforts to control TMs have been effective in reducing the number of outstanding TMs.

No violations or deviations were identified in the areas inspected.

7. Action on Previous Inspection Findings (92701)

a. (Open) IFI 50-321,366/89-08-03, PSW System Design Pressure.

A question was raised during the SSFI with respect to pressure experienced by various pieces of equipment in the Reactor Building which are serviced by Plant Service Water System. Specifically, during an accident wherein the Turbine Building plant service water flow isolates, the PSW pump backs up its performance curve, resulting in system pressure slightly higher than normal. The piping specifications cover the higher pressure by specifying design and maximum pressure at 180 and 190 psig, respectively. However, several components have been identified as having a design pressure lower than that expected during an accident.

The licensee's action, upon discovery of this item, included a justification for continued operations and a commitment to resolve this issue in the long term. The JCO was reviewed and accepted by the NRC and work has progressed on the long-term fix. The long-term fix includes several modifications. Design change packages have been prepared and approved, but the installation has not been done. Final resolution of this issue will be completed when all hardware fixes are installed. These are scheduled for future planned outages.

b. (Closed) IFI 50-321,366/89-08-04, Common 10-Inch PSW Discharge Line.

This issue relates to a safety concern raised during the SSFI regarding the common cooling water discharge line for the EDG and the possibility of losing the cooling water for the EDG due to line blockage.

The licensee responded to this question by reviewing the drawings of the piping system (following the inspection). The design drawings, H-11600 and H-11146, show a rupture disc installed downstream of all branch lines connecting to the common discharge header. The rupture disc is designed to fail at 95 psig, which is a realistic pressure that would be experienced if such a catastrophic failure occurred.

In addition, the licensee performed a probabilistic evaluation to determine the risk significance of this failure. The evaluation concluded that pipe collapse contributes less than one percent to the frequency of station blackout occurring at Hatch due to all causes.

Therefore, considering the above two actions, this issue is closed.

c. (Closed) IFI 50-321,366/89-08-07, Emergency Diesel Generator CARDOX System.

During the SSFI, the design basis of the EDG CARDOX system was questioned because the basis did not consider the ability of the fire detector switches to withstand a seismic event. In response to this issue, the licensee had the components in question tested at the Wyle Laboratories. All components tested met the established criteria for Plant Hatch.

Wyle submitted a report, Reference No. 41033B-002, to Southern Company Services, dated April 17, 1990. The test report described the test requirements, procedures, and results. The test was completed with no abnormalities and concludes that inadvertent activation of the CARDOX system would not occur during a seismic event. This closes all action on this issue.

d. (Closed) IFI 50-321,366/89-08-08, Seismic Qualification of EDG Low Lubricating Oil Switches.

The status of this item was reported in NRC Inspection Report No. 89-30 and was kept open pending receipt of the seismic qualification documentation for the Allen Bradley pressure switches. These switches are Allen Bradley type 836-C2, MPL No. R43-N757A and R45-N:757C. To seismically qualify the switches, the licensee sent samples of ANCO Engineering, Inc., for testing. The test report indicated the test conditions and concluded that no chatter or bounce occurred upon change of switch state during the seismic test. Based on these results, this issue is closed.

e. (Closed) IFI 50-321,366/89-08-10, Electro-thermo Links on Diesel Room Roll-up Doors and Fire Dampers.

The status of this item was reported in NRC Inspection Report No. 90-11. The item was kept open because no qualification documentation was available for the fire damper electrothermal fusible links. The roll-up doors and fire dampers for the EDG building are not safety-related and were purchased without seismic qualification. The electrothermal links are a component of the roll-up doors and fire dampers.

Seismic calculations for the fusible links have been completed by Southern Company Services. It was determined that the fusible links are seismically adequate for the installation at Hatch. Therefore, the fusible link should not fail during a seismic event, and the doors should remain open to perform their intended task. The calculations reviewed were SCN 89-029, Revision 0, and SCN 89-030, Revision 0. This completes all action required for this item.

8. Exit Interview (30703)

The inspection scope and results were summarized on October 19, 1990, with those persons indicated in paragraph 1. The inspectors described the areas inspected and discussed in detail the inspection results listed below. Proprietary information is not contained in this report. Dissenting comments were not received from the licensee.

The licensee was informed that the following items were closed.

IFI 50-321,366/89-08-04, Common Ten Inches PSW Discharge Line

IFI 50-321,366/89-08-07, Emergency Diesel Generator CARDOX System

IFI 50-321,366/89-08-08, Seismic Qualification of EDG Low Lubricating Oil Switches

IFI 50-321,366/89-08-10, Electro-thermo Links on Diesel Room Roll-up Doors and Fire Dampers

The licensee was informed that the following item remains open.

IFI 50-321,366/89-08-03, PSW System Design Pressure

9. Acronyms and Initialisms

ABN	As Built Notice
DC	Deficiency Card
DCP	Design Change Package
DCR	Design Change Request
EDG	Emergency Diesel Generator
FCR	Field Change Request
GE	General Electric
IFI	Inspector Followup Item
JCO	Justification for Continued Operation
LER	Licensee Event Report
LPCI	Low Pressure Coolant Injection
MWO	Maintenance Work Order
NRC	Nuclear Regulatory Commission
OEM	Original Equipment Manufacturer
PO	Purchase Order
PRB	Plant Review Board
PSIG	Pounds Per Square Inch Gauge
PSW	Plant Service Water
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RER	Request for Engineering Review
SCS	Southern Company Services
SIL	Service Instruction Letter
SOR	Significant Occurrence Report
SSFI	Safety System Functional Inspection
STA	Shift Technical Advisor
TM	Temporary Modification
TS	Technical Specification
USQ	Unreviewed Safety Question