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

Report Nos.: 50-348/92-16 and 50-364/92-16

Licensee: Southern Nuclear Operating Company, Inc.
600 North 18th Street
Birmingham, AL 35291-0400

Docket No.: 50-348 and 50-364 License No.: NPF-2 and NPF-8

Facility Name: Farley 1 and 2

Inspection Conducted: May 18-22, 1992

Inspector: C. C. Smith 6-25-92
C. Smith Date Signed

for Frank Jape 6/25/92
M. Thomas Date Signed

Laurence P King 6/25/92
L. King Date Signed

Approved by: Frank Jape 6/25/92
F. Jape, Chief Date Signed
Test Programs Section
Engineering Branch
Division of Reactor Safety

SUMMARY

Scope:

This routine, announced inspection was conducted in the areas of design, design change and plant modification including engineering technical support.

Results:

In the areas inspected, two violations were identified, failure to incorporate setpoint tolerances into plant drawings, (paragraph 2.b) and failure to specify and perform an adequate post modification test of service water flow, (paragraph 3.c).

REPORT DETAILS

1. Persons Contacted

Licensee Employees

*W. Bayne, Supervisor, Safety Audit and Engineering Review
*R. Coleman, Manager, Plant Modifications Department
A. Davidson, Supervisor, Configuration Management Support Section
*S. Casey, System Performance Supervisor
L. Enfinger, Manager, Plant Administration
*R. Hill, General Manager, Nuclear Plant
*C. Nesbitt, Manager, Operations
J. Odom, Superintendent, Unit Operations
J. Powell, Superintendent, Unit Operations
*L. Stinson, Assistant General Manager, Operations
*J. Thomas, Manager, Operations
R. Tyler, Supervisor, Maintenance Engineering Support
R. Winkler, Supervisor, Plant Modifications
R. Woodfin, Supervisor, Maintenance and Operations Support Section

Other licensee employees contacted during this inspection included craftsmen, engineers, operators, mechanics, technicians, and administrative personnel.

NRC Personnel

*F. Jape, Test Programs Section Chief
*G. Maxwell, Senior Resident Inspector
M. Morgan, Resident Inspector

*Attended exit interview

2. Design Changes and Plant Modifications (37700)

a. Plant Modifications to Improve Reactor Safety

An evaluation of the licensee's initiatives taken to identify and implement plant modifications to improve reactor safety was performed during this inspection. Objective evidence reviewed during this effort included:

- Unit 2 Cycle 8-9 Design Change Summary
- Unit 1 Cycle 10-11 Design Change Summary

- Unit 1 Proposed 11 RFO Plant Modifications List
- Operations and Maintenance and Capital Budgets 1992-1995
- Procedure No. FNP-O-AP-70, Conduct of Operations, Plant Modifications and Maintenance Support Group, Revision 2

The inspectors determined that responsibility has been assigned to the Plant Modifications Manager for completing a budget classification for plant modifications which will be added to the design change Work List Revision (WLR). An integral part of the budget classification process involves the use of the WLR Priority system where the proposed work is prioritized in accordance with weighting factors that are related to the nuclear safety significance of the station problem. The inspectors reviewed procedure FNP-O-AP-70 and determined that the budget classification process had not been addressed. In discussions with licensee management the inspectors were informed that the prioritization process is a management tool which involves the collective input of all plant managers for its use. Every six months the design change WLR is reviewed by licensee management who collectively agree on the priority assigned to proposed plant modifications. Allocation of funds for development of the PCNs are based on the consensus reached concerning the priority of the proposed plant modification. Additionally, the scheduled dates for installing the proposed plant modifications are commensurate with the safety significance of the work to be performed.

The inspectors reviewed the listed objective evidence and concluded that licensee management had demonstrated the use of a prioritization process for identifying and implementing plant modifications. The increasing order of importance of the PCR worklist priority numbers is based on activities which ranges from convenience improvements to personnel/reactor safety. Inspection of design change packages having priority numbers and the associated scope of the design changes further demonstrate the use of the priority numbers for resolving issues from a nuclear safety standpoint. The inspectors performed independent design reviews of 6 PCNs having worklist priority numbers ranging from 4.487 to 1.240 during this effort. Additional examples of the licensee's use of a prioritization process for resolving station problems is found in the DBD Program which is addressed in paragraph 2.c.

Licensee's activities related to initiatives taken for identifying and implementing plant modifications to improve reactor safety was identified as a strength.

b. Planning, Development and Implementation of Plant Modifications

The inspectors reviewed the PCNs listed below to determine the adequacy of the evaluations performed to meet 10 CFR 50.59 requirements; verify that the PCNs were reviewed and approved in accordance with TS and administrative controls; ensure the subject modifications were installed (for those physically inspectable) in accordance with the PCN packages; applicable plant operating documents (drawings, plant procedures, FSAR, TS, etc.) were revised to reflect the subject modifications the modifications were reviewed and incorporated in operations training programs as applicable; and post modification test requirements were specified and adequate testing performed.

° PCN # 92-2-8108, Containment Cooler Breaker Settings.

This PCN provided design basis information that was used to revise relays settings of the 600 Volt load center breakers EA-10 and ED-15. This station problem was caused by Containment Cooler 2A replacement motor having a higher value of locked rotor current than the original motor. The 10 CFR 50.59 Nuclear Safety Screening bounded the activities specified in the design scope and was determined to be technically adequate.

° PCN # 91-2-7749, Access Hatch for Reactor Vessel Head Vent Valve Junction Box

This PCN modified the cable support platform to allow access to a junction box without having to remove the cable support platform or the installation of scaffolding. This PCN was assigned a WLR Priority No. of 2.105 which is related to personnel radiological safety. No deficiencies were identified with the PCN or the 10 CFR 50.59 Nuclear Safety Screening.

° PCN # 91-1-7841, SBO Motor Driven Fire Pump Auto-start Interlock

The scope of the above plant modification involved (1) modifying the motor driven fire-pump control circuitry to prevent automatic starting of the pump when offsite power is not available; (2) rewire pressure switch N1P43PS508 in series with the normally closed contact of the push-button; and (3) adding a class 1E fuse to isolate the non-class 1E circuits from the class 1E. Review of the design change package verified that the 10 CFR 50.59 Nuclear Safety Screening bounded the activities within the design change scope and was technically adequate. No deficiencies were identified.

° PCN # B87--1-4118, Relocation of Primary Platform Ladder

This PCN relocated the primary platform from the north side of steam generator "A" and eliminated the radiological hazard to the workers caused by passing within six feet of the regenerative heat exchangers. The design objective of reducing workers doses was accomplished by this PCN which was assigned a WLR Priority No. 2.025. No deficiencies were identified.

° PCN # 88-1-5247, Upgrade Lead Blanket Shielding Frames for Permanent Installation in Containment

This PCN was assigned a WLR Priority No. 2.027 and had as a design objective the reduction of workers doses caused by the assembly and dis-assembly and removal of the lead shielding frames from the containment each outage. The scope of the design change involved modifying the lead shielding frames to meet structural requirements that satisfies seismic qualification criteria and which permitted the frames to be left in the containment with the reactor at power. A seismic analysis was performed per calculation C863917 and the 10 CFR 50.59 Nuclear Safety Screening adequately bounded activities within the design change scope. No deficiencies were identified.

° PCN # 89-1-5586, Lonergan Relief Valve Spring Replacement

Engineering Safety Report EQ-88-1185, identified a station problem with various Lonergan Model LCT-1 pressure relief valves that had set-point pressure values which exceeded the range of the installed springs, i.e. 96-145 psig. The above PCN was developed and implemented to (1) replace these springs with springs having a range of 146-220 psig and (2) to revise the Instrument Setpoint Index, Drawing No. B-175968, to show setpoint tolerances of ±4.5 psi for these valves. The design scope identified a total of 43 pressure relief valves that were used within the Component Cooling Water, and Service Water Systems. Review of the design change package revealed that the installation of the new springs were completed in September 1989. However, The Setpoint Index was not revised, to document the design basis information concerning setpoint tolerances that is required by plant staff to ensure proper calibration of the relief valves. This finding was identified as Violation 50-348/92-16-01, Failure of design controls to ensure incorporation of setpoint tolerances into plant drawings.

The inspectors concluded that the quality and technical content of the design input information for the above PCNs was good. The design scope clearly identified the activities required to achieve design objectives, and where required, post-modification test requirements and test acceptance criteria were specified for the PCNs reviewed. The 10 CFR 50.59 Nuclear Safety Screening were considered technically adequate and bounded the activities within the design change scope.

c. Minor Departures

Ten minor departures were reviewed and no problems were found.

d. Configuration Management Program

An evaluation of the licensee's activities involving configuration of a Management Program was performed during this inspection. Objective evidence reviewed during this effort included the following:

- Project Plan Configuration Management Program, Farley Nuclear Plant, APCo, Revision 0
- Procedure for Preparation of Functional System Descriptions, Farley Nuclear Plant, APCo, Revision 2
- Farley Two Year Capital Budget Plan, 1992-93
- Farley Nuclear Plant Four Year O/M Budget Plan 1992-95 Funding

The inspectors determined that the licensee had developed a Configuration Management Program based upon the results of a self initiated self-assessment of the Service Water System. The Configuration Management Program administrative controls are contained in the Project Plan. Paragraph 2.0 of the Project Plan addressed the scope of the program and identified the components of the plan which are related to various DBD activities. Among these activities are development and use of FSDs; development and use of "Q" List; consolidated setpoint documents and development of a Configuration Management (CM) Manual. These activities are not all inclusive the scope of activities in the CM program.

The licensee is presently developing and using the FSDs. Responsibilities have been assigned and estimated cost and schedules have been developed for a five year CM Program. A total of 13 systems have been identified for inclusion in the program. To date the licensee has completed and issued FSD for the Component Cooling Water System; the Service Water Systems; RHR/LHSI System; Containment Isolation System; the Electrical Distribution System and the Diesel Generators. The licensee evaluates the technical adequacy of the FSDs and their conformance with plant programs by performing a self initiated safety system assessment (SSSA) upon completion of each FSD. Discrepancies or open items identified during the SSSAs are dispositioned by use of a prioritization process which ensures resolution of problems from a nuclear safety standpoint.

The inspectors reviewed the list of PCNs prepared as a result of the licensee's SSSA and verified implementation of the prioritization of station problems. Licensee's initial processing of station problems identified via SSSA involve logging, prioritizing, and review for reportability requirements. The licensee's activities related to the development and implementation of the CM Program was identified as a strength.

3. Engineering and Technical Support

The inspectors reviewed organization and staffing and the activities of various plant groups in an effort to assess the timeliness and effectiveness of the engineering support provided to plant operations and maintenance staffs for day-to-day plant activities. This included activities of the Systems Performance Group and the Maintenance Engineering Support Group.

The inspectors concluded from reviewing the activities of these groups that, in general, timely and effective support was being provided to the plant. There were examples where the support was considered good and there were examples where the support was considered less than adequate. Examples are discussed in greater detail in the following paragraphs.

a. Organization and Staff

Engineering and technical support was provided by onsite and offsite engineering organizations. Onsite support was provided primarily by the Systems Performance Group, Maintenance Engineering Support Group, Plant Modifications Department, Technical Department, and the Operations Support Group. Onsite engineering and technical support was generally provided for items that were small in scope. More complex and larger scope items were referred to the corporate engineering organizations who provided the necessary support. The inspectors concluded that the onsite engineering and technical support groups were adequately staffed with knowledgeable personnel.

b. Problem Identification and Resolution

The inspectors assessed engineering involvement in problem identification and resolution activities. Involvement was assessed by reviewing support requests and problem reports of the Systems Performance Group (SP); evaluation reports of the Maintenance Engineering Support Group (MESG); and incident reports of the Plant Modifications Department (PMD). Selected reports assigned to the above groups were reviewed for 1991 and 1992. The inspectors concluded from reviewing the above documents that, except for a few instances, licensee engineers identified and resolved technical issues for both operations and maintenance. The resolutions were generally adequate.

c. Support Requests and Problem Reports

The inspectors reviewed selected support requests and problem reports assigned to the SP group during 1991 and 1992. Licensee personnel stated that problem reports were generally used by SP as the mechanism for responding to a support request from other plant groups. There were a few instances where a support request was answered without a problem report being generated, but in most cases, for each support request there was a corresponding problem report.

During review of support request MOS-91.018, Service Water Flow to 1C Diesel Generator, the inspectors noted that the request was written for SP to measure the service water (SW) flow to the 1C diesel generator (D/G) per maintenance work request (MWR) MWR 241708 using a Controlotron ultrasonic flow meter. Operations had received an alarm indicating a low SW flow while performing surveillance test procedure FNP-O-STP-80.1, Diesel Generator 1-2A Operability Test for D/G 1-2A on July 12, 1991. MWR 240105 was written to investigate the problem for D/G 1-2A. MWR 240105 was completed for D/G 1-2A on August 22, 1991 and it stated that SW flow to D/G 1-2A was adequate and no problems were found. MWR 241708 was completed October 25, 1991 when SP measured the SW flow to D/G 1C while operations performed FNP-O-STP-80.2, Diesel Generator 1C (2C) Operability Test. The acceptance criteria specified in the MWR was a minimum SW flow of 950 gallons per minute (gpm) to D/G 1C. The test results showed that with Unit 2 only supplying SW to the 1C D/G, only 640 gpm flow was measured. This was less than the 950 gpm specified in the acceptance criteria which was based on the design basis case of a postulated loss of the river water system and the SW temperature rising to greater than 96 degrees Fahrenheit (F), up to a maximum temperature of 110 degrees F. Systems Performance recommended that the test data obtained for the condition with only Unit 2 supplying SW to the 1C D/G be transmitted to the corporate Nuclear Support Department and an evaluation be performed based on the SW flow balance and recommendations be provided to correct the problem. Operations issued an attachment to the Night Order Book on October 25, 1992, stating that the flow measured to D/G 1C was acceptable. The inspectors discussed this item with operations personnel during the inspection who stated that their conclusion was based on the condition where SW temperature was less than 96 degrees F and the required SW flow to D/G 1C was 540 gpm. Operations personnel further stated that the SW temperature was less then 96 degrees F when the flow test was run in October 1991.

The inspectors discussed the item further with SP personnel who stated that the data was never sent to corporate Nuclear Support for evaluation after Corporate Nuclear Maintenance Support notified the site on October 29, 1991, that a section of the SW return piping from D/G 1C was scheduled for radiography testing (RT) prior to the Unit 2 refueling outage (RFO) in order to determine the amount of blockage in the piping. The D/G 1C piping was selected for RT as part of the licensee's SW Piping RT Inspection Program. This program was independent of the flow testing performed by SP. Corporate Nuclear Maintenance Support did not appear to be aware of the flow testing performed by SP and the associated low flow concern. There was no mention of the testing performed nor the low flow concern in the October 29, 1991 memo from Corporate Nuclear Maintenance Support to the plant. Also, the testing and low flow concern were not mentioned in Revision 28 to PCN B-87-2-4106. This PCN revision was written to replace the D/G 1C SW return piping after the piping was examined by RT in December 1991 and found to have approximately 40 percent blockage. The piping was replaced during the 1992 Unit 2 RFO.

The inspectors reviewed revision 28 of PCN B-87-2-4106. Areas reviewed included but were not limited to those stated in paragraph 2.b. of this inspection report. The inspectors verified that various design inputs such as material, temperature, pressure, seismic, stress, pipe supports, structural integrity, as well as other design inputs were reviewed and evaluated. During further review of Revision 28 to determine post modification test requirements, the inspectors noted that there was no requirement to verify that the SW design flows to D/G 1C could be achieved. The inspectors questioned licensee personnel concerning the post modification test requirements who stated that the hydrostatic tests performed to verify the integrity of the welds was considered adequate testing for the scope of the modification. The inspectors stated that performing hydrostatic testing only was not considered adequate post modification testing for verifying that adequate SW flow to D/G 1C could be achieved for the design basis conditions specified. This concern was particularly noteworthy since it had been previously determined through testing that adequate SW flow to D/G 1C could not be achieved for all design basis conditions. Also, summer is approaching and the SW temperature will increase. The inspectors also expressed concern that there was no requirement to verify that SW flow to other components was not affected by the modification. The inspectors informed the licensee that the failure to specify and perform adequate post modification testing for revision 28 to PCN B-87-2-4106 would be identified as a violation of 10 CFR 50, Appendix B, Criterion III. This item will be identified as 50-364/92-16-02.

During review of support request OEE-91.024, Diesel Generator 1C Response with 1-2A Unavailable, the inspectors noted that problem report PR-130 was also written. The support request and problem report stated that if D/G 1-2A were not available with its mode selector switch in mode 3, and a dual unit loss of offsite power occurred, then there would be no source of "A" train power. D/G 1C would be running with no "A" train SW pumps in operation to provide cooling water D/G 1C. The plant requested support from corporate who in turn requested support from the architect engineer to resolve the issue. SP recommended some short term and long term actions based on the evaluation performed by the architect engineer. These recommendations included changes to the D/G system operating procedures to ensure D/G 1C would not be affected when D/G 1-2A is tagged out, and a design change to the diesel logic so that D/G 1C operation would not be dependent on the position of D/G 1-2A mode selector switch. Procedure SOP-38, Diesel Generator Operation was in the process of being revised to incorporate the recommendations from SP. In addition, SP had also initiated PCR 92-1-8143 for Unit 1 and PCR 92-2-8144 for Unit 2 to change D/G 1C alignment and loading logic. The inspectors considered the licensee's efforts to resolve this issue to be a good example of engineering support provided to the plant.

The inspectors reviewed aspects of the licensee's Service Water Upgrade Program which include the Service Water Inspection Program and the performance monitoring of SW flow to various safety related components. The SW inspection program was developed to detect and quantify flow area reduction and pipe wall thinning in SW piping through the use of RT. It was through the SW RT inspection program that the licensee identified the need to replace the SW return line from D/G 1C. The licensee developed a program to monitor SW flow supplied to various safety related components in response to NRC Generic Letter 89-13. SW flow is monitored to components such as the D/Gs, the containment air coolers, and various safety related room coolers. The inspectors considered the licensee's Service Water Upgrade Program to be a good example of engineering support.

d. Maintenance Engineering Support Group

The Maintenance Engineering Support Group (MESG) was organized into five functional areas with each functional area reporting to the Supervisor, MESG. These groups include support engineering, procedure review and development, valve support, predictive maintenance and generating plant technical services. The inspectors reviewed the functions of the reliability centered maintenance group which is a part of the procedures review and development group and the predictive maintenance group.

(1) Reliability Centered Maintenance

The inspectors reviewed progress on the reliability centered maintenance computer program. The program is part of MESG and was staffed with a supervisor and three maintenance foreman. The three systems completed were the service water, residual heat removal and component cooling water system. Presently the group is inputing information for the waste gas, instrument air and cooling tower fans.

A weakness in the system at present is that it does not contain model numbers for all instruments. Therefore, if a problem was found in a instrument in one system, the licensee would not be able to determine all places it is used.

(2) Predictive Maintenance

The inspectors reviewed the predictive maintenance program which is part of MESG. The program included vibration analysis, oil analysis and thermography. This was accomplished by reviewing engineering evaluation reports written over the SALP period. Reports were written where recommendations were required. Most of the recommendations that were made resulted in increased vibration monitoring and oil changeouts. In a few cases, there was no documentation of follow-up although the licensee indicated it had been accomplished.

c. Configuration Control Group

A significant effort has been made to develop a program for trending equipment failure. Three contractors went though past work orders to code the completion checklist to ensure information could be extracted from completed work requests.

A program was developed by Southern Services to extract historical information from the work requests for equipment trending.

The inspectors reviewed the training procedures for maintenance foreman on how to fill out the work request checklist and found it adequate.

The equipment trending program exceeds the requirements of NPRDS and covers all equipment at the plant.

4. Exit Interview

The inspection scope and results were summarized on May 22, 1992, with those persons indicated in paragraph 1. The inspector described the areas inspected and discussed in detail the inspection results listed below. Proprietary information is not contained in this report.

Two violations were identified and are described as follows:

- 50-348/92-16-01, Failure of the design controls to ensure incorporation of setpoint tolerances into plant drawing (paragraph 2.b)

- 50-364/92-16-02, Failure to specify and perform adequate post modification testing (paragraph 3.c)

Dissenting comments were received from the licensee concerning the second violation. Licensee personnel stated that the hydrostatic testing was considered adequate for the scope of work performed, but they would consider the issue of lack of flow testing upon receipt of the report.

5. Acronyms and Initialisms

WLR	Work List Revision
RFO	Refueling Outage
PCR	Production Change Request
PCNs	Production Change Notice
TS	Technical Specification
FSAR	Final Safety Analysis Report
DBD	Design Basis Document
APCo.	Alabama Power Company
FSD	Functional System Description
RHR/LHSI	Residual Heat Removal/Low Head Safety Injection
RFO	Refueling Outage
CM	Configuration Management
SP	Systems Performance Group
MESG	Maintenance Engineering Support Group
PMD	Plant Modifications Department
D/G	Emergency Diesel Generator
SW	Service Water
MWR	Maintenance Work Request
NPRDS	Nuclear Plant Reliability Data System
RT	Radiographs Test