UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
61 FORSYTH STREET SW SUITE 23T85
ATLANTA, GEORGIA 30303-8931

March 23, 2000

EA 98-467

Southern Nuclear Operating Company, Inc.
ATTN: Mr. D. N. Morey
Vice President
P. O. Box 1295
Birmingham, AL 35201-1295

SUBJECT: NRC INTEGRATED INSPECTION REPORT NOS. 50-348/00-01 and 50-364/00-01

Dear Mr. Morey:

On February 26, 2000, the NRC completed an inspection at your Farley Nuclear Plant. The enclosed integrated report presents the results of that inspection. During the five-week period covered by this inspection, your conduct of activities at the Farley Plant facilities was generally characterized by safety-conscious operations and appropriate maintenance and plant support activities.

Based on the results of this inspection, the NRC has determined that a violation of NRC requirements occurred. This violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1.a of the Enforcement Policy. This NCV is described in the subject inspection report. If you contest the violation or the severity of this NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001, with copies to the Regional Administrator, Region II, and the Director, Office of Enforcement, USNRC, Washington, D.C. 20555-0001.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be placed in the NRC Public Document Room (PDR).

Sincerely,

(Original signed by)
Stephen J. Cahill, Chief
Reactor Projects, Branch 2
Division of Reactor Projects

Docket Nos. 50-348 and 50-364
License Nos. NPF-2 and NPF-8

Enclosure: NRC Inspection Report Nos. 50-348/00-01
and 50-364/00-01

cc w/encl: (See Page 2)

cc w/encl:
M. J. Ajluni, Licensing
Services Manager, B-031
Southern Nuclear Operating
Company, Inc.
Electronic Mail Distribution

L. M. Stinson
General Manager, Farley Plant
Southern Nuclear Operating
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Rebecca V. Badham
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M. Padovan, NRR
PUBLIC

OFFICE	RII:DRP	RII:DRP	RII:DRP	RII:DRP	RII:DRS	RII:DRS	RII:DRS
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NAME	CWRapp:sjw	TPJohnson	RKCaldwell	JBartley	DForbes		
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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-348 and 50-364
License Nos.:	NPF-2 and NPF-8
Report Nos.:	50-348/00-01 and 50-364/00-01
Licensee:	Southern Nuclear Operating Company, Inc.
Facility:	Farley Nuclear Plant, Units 1 and 2
Location:	7388 N. State Highway 95 Columbia, AL 36319
Dates:	January 23 to February 26, 2000
Inspectors:	T. P. Johnson, Senior Resident Inspector R. K. Caldwell, Resident Inspector J. H. Bartley, Resident Inspector D. B. Forbes, Radiation Specialist (Section R1.2)
Approved by:	Stephen J. Cahill, Chief Reactor Projects, Branch 2 Division of Reactor Projects

EXECUTIVE SUMMARY

Farley Nuclear Power Plant Units 1 and 2
Nuclear Regulatory Commission Inspection Report 50-348,364/00-01

This integrated inspection to assure public health and safety included aspects of licensee operations, maintenance, engineering, and plant support. The report covers a five-week period of resident inspections and a specialist inspection of radiation protection for the steam generator replacement project.

Operations

- Operator performance during normal unit operations and during Unit 1 operation at reduced power due to an extraction steam leak in the main condenser was appropriate (Section O1.1).

Engineering

- A non-cited violation was identified for the failure to identify and submit a license amendment request for an unreviewed safety question for elevated reactor vessel support concrete temperatures (Section E8.1).

Plant Support

- Radiation Protection and As Low As Reasonably Achievable preplanning for the upcoming steam generator replacement outages was progressing satisfactorily (Section R1.2).

- A Chemistry Department self-assessment was noted to be self-critical and thorough based on the depth of the audit, the licensee's findings, and proposed corrective actions (Section R7.1).

- Fire brigade response to a fire outside the protected area was prompt and effective (Section F4.1).

REPORT DETAILS

Summary of Plant Status

Unit 1 operated at or near Rated Thermal Power (RTP) until January 24, when power was reduced to 62% RTP due to an extraction steam leak in the main condenser.

Unit 2 operated at or near RTP power for the period.

I. Operations

O1 Conduct of Operations

O1.1 Routine Observations of Control Room and Related Operations (60705 and 71707)

The inspectors observed licensed control room operator and non-licensed operator performance during the period. Operator attentiveness to annunciator alarms and response to changing plant conditions were prompt. Operator performance was appropriate during Unit 1 operation at reduced power due to an extraction steam leak in the main condenser.

O2 Operational Status of Facilities and Equipment

O2.1 General Tours and Inspections of Safety Systems (71707)

General tours of safety-related areas were performed by the inspectors to observe the physical condition of plant equipment and structures, and to verify that safety and risk significant systems were properly maintained and aligned. Plant systems inspected included service water, Auxiliary Feedwater (AFW), Component Cooling Water (CCW), Containment Spray (CS), and Emergency Diesel Generator (EDG). The inspectors also verified that selected tagouts were implemented in accordance with procedural requirements. During these general tours, the inspectors noted differences in oil bubbler orientation on selected CCW, AFW, and CS pumps. The licensee contacted the vendor and determined that the oil bubblers were misoriented but the pumps were not inoperable. The licensee initiated corrective actions to properly orient the oil bubblers. The inspector reviewed the licensee actions and concluded that the licensee's response to the concerns was appropriate.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments (50001, 61726, and 62707)

The inspectors witnessed or reviewed portions of selected maintenance and surveillance test activities. This included preparations for the steam generator replacement (SGRP) outage, Reactor Protection System testing, Service Water Booster Lube and Cooling pump maintenance, 1C EDG testing, and AFW pump maintenance. The inspectors determined that the activities observed were conducted in a satisfactory manner and that the work was properly performed in accordance with approved maintenance work orders. Personnel conducting the activities were knowledgeable of their assigned tasks. Related equipment tagouts were reviewed and determined to be adequate.

III. Engineering

E8 Miscellaneous Engineering Issues

E8.1 (Closed) EEI 50-348, 364/98-05-02, Failure to Identify Defacto 50.59 and USQ (92903)

a. Inspection Scope (92903)

The inspectors reviewed Escalated Enforcement Item (EEI) 98-05-02 and the NRC staff's response to Task Interface Agreement (TIA) 98-11, "Farley's Interpretation of ACI Code for Reactor Vessel Support Concrete Temperatures."

b. Observations and Findings

EEI 98-05-02

As documented in IR 50-348, 364/98-05, in 1977 and again in 1997, the licensee identified that the Reactor Vessel Support (RVS) concrete temperature was greater than the temperature stated in the Updated Final Safety Analysis Report (UFSAR). On December 18, 1997, the licensee approved a change to the UFSAR for the elevated RVS concrete temperatures based on the American Concrete Institute (ACI) codes. The inspectors documented that the licensee did not identify that RVS concrete temperature greater than the temperature stated in the UFSAR was potentially an unreviewed safety question (USQ). As documented in the response to TIA 98-11, the staff concluded that the licensee was not properly applying the ACI codes. The degradation of the RVS concrete as a result of the increased temperature could result in an increase of the RVS malfunction probability as previously evaluated in the UFSAR. 10 CFR 50.59(c) requires, in part, the holder of a license who desires to make a change in the facility or the procedures described in the safety analysis report which involve an unreviewed safety question, shall submit an application for amendment of his license pursuant to 50.90. 10 CFR 50.59 (a)(2) states, in part, a proposed change shall be deemed to involve an unreviewed safety question if the probability of occurrence an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased.

Contrary to the above, on December 18, 1997, the licensee approved a change to the UFSAR which failed to identify that allowing higher RVS concrete temperatures was an unreviewed safety question and did not submit an application to amend the license. This NRC identified Severity Level IV violation is being treated as a Non-Cited Violation (NCV) consistent with Section Section VII.B.1.a of the NRC Enforcement Policy. This violation is identified as NCV 50-348, 364/00-01-01, "Failure to Identify an USQ" and is in the licensee's corrective action program as Occurrence Report (OR) 1-2000-098.

UFSAR Discrepancies

Also discussed in IR 50-348, 364/98-05, were two potential UFSAR discrepancies that could impact assumptions for the RVS concrete temperature thermal analysis. Based on further inspector reviews, the discrepancy concerning location of temperature detectors could be changed without requiring NRC approval. The second discrepancy concerning air flow path was adequately described in the UFSAR. The inspectors also concluded that neither of these items invalidated the inlet air temperature assumption of 120°F.

c. Conclusions

A non-cited violation was identified for the failure to identify and submit an amendment request for an unreviewed safety question for elevated reactor vessel support concrete temperatures.

E8.2 Licensee Event Report (LER) 50-348/98-08-00: Reactor Vessel Suport Concrete Design Basis Temperature Exceeded Due to Closed Cooling Damper

The inspectors reviewed the licensee's conclusions in this previously closed LER with respect to the staff's conclusions in the response to TIA 98-11. In the LER, the licensee concludes that temperatures as high as 300°F would not degrade the RVS concrete. This is contrary to the staff's conclusion that the RVS concrete could degrade at temperatures above 190°F. The licensee was appropriately evaluating this discrepancy as part of OR 1-2000-098.

IV. Plant Support

R1 Radiological Protection and Chemistry Controls

R1.1 Radiologically Controlled Area (RCA) Tour and Radiation Exposure (71750)

Overall control of material and activities of the RCA remained effective. Plant personnel observed working in the RCA demonstrated appropriate knowledge and application of radiological control practices. Health physics technicians provided positive control and support of work activities in the RCA.

R1.2 Steam Generator Replacement Preplanning Inspection (50001)

The inspectors toured the Old Steam Generator Storage Facility and reviewed the radiological aspects of facility design with Radiation Protection (RP) management. The storage facility was constructed for weather protection for the steam generators that were to be retired, as well as external building dose rate reduction.

The inspectors reviewed containment shielding plans for higher dose work and observed facility preparations for the upcoming Units 1 and 2 SGRP outages. The inspectors also discussed As Low As Reasonably Achievable (ALARA) preparations with RP management. The licensee had incorporated RP and ALARA lessons learned from other licensees during the preplanning of SGRP work. The inspectors observed existing mockup training facilities and reviewed planned mockup training for higher dose work. To incorporate lessons learned during the Unit 1 SGRP, the licensee planned to use basically the same RP staff and ALARA coordinators for the Unit 2 SGRP outage. The inspectors determined RP and ALARA preplanning for the SRGP outages was progressing satisfactorily.

R7 Quality Assurance in RP&C Activities

R7.1 Chemistry Self-Assessment (71750)

The inspectors reviewed a Chemistry Department self-assessment that was conducted by both onsite and offsite personnel. The assessment team appropriately presented management with their findings and recommended corrective actions in each area. Recommendations included comments on upgrading the cleanliness and material condition of the laboratories and sampling rooms to improvements in procedures. A final report was being processed and licensee management was considering appropriate corrective actions. Based on the depth of the audit, the licensee's findings, and proposed corrective actions, the inspector concluded the assessment was self-critical and thorough.

P1 Conduct of EP Activities

P1.1 Emergency Drill (71750)

The inspectors observed portions of a quarterly EP drill that was conducted on February 10. Activities at the onsite emergency response facilities were observed. The drill was conducted well and no significant issues were identified.

F4 Fire Protection Staff Knowledge and Performance

F4.1 Grass Fire Outside Protected Area (71750)

On February 7, the resident inspectors observed the licensee response to a grass fire inside the Owner Controlled Area, but outside the nuclear plant protected area. The fire brigade responded to the fire location properly dressed out and had water on the fire within 15 minutes. The fire was extinguished within 30 minutes of the initial notification. No off-site assistance was required. The inspectors concluded the fire brigade was well trained and responded promptly to the fire. Because the fire was outside the Protected Area, the licensee appropriately determined the fire was not reportable.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on February 28, 2000. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

Partial List of Persons Contacted

Licensee

R. V. Badham, Safety Audit Engineering Review Supervisor
C. L. Buck, Technical Manager
R. M. Coleman, Outage and Modification Manager
C. D. Collins, Operations Manager
K. C. Dyar, Security Manager
S. Fulmer, Plant Training and Emergency Preparadness Manager
J. S. Gates, Administration Manager
D. E. Grissette, Assistant General Manager - Operations
J. G. Horn, Outage Planning Supervisor
J. R. Johnson, Maintenance Manager
R. R. Martin, Engineering Support Manager
C. D. Nesbitt, Assistant General Manager - Plant Support
L. M. Stinson, Plant General Manager - FNP
R. J. Vanderbye, Emergency Prepardness Coordinator

List of Opened, Closed, and Discussed Items

Type	Item Number	Description and Reference
Closed		
NCV	50-348, 364/00-01-01	Failure to Identify an USQ (Section E8.1)
EEI	50-348, 364/98-05-02	Failure to Identify Defacto 50.59 and USQ (Section E8.1)
Discussed		
LER	50-348/98-08-00	Reactor Vessel Support Concrete Design Basis Temperature Exceeded Due to Closed Cooling Damper (Section E8.2)

List of Inspection Procedures (IP) Used

IP 37551: Onsite Engineering
IP 50001: Steam Generator Replacement
IP 60705: Preparation for Refueling
IP 61726: Surveillance Observations
IP 62707: Maintenance Observations
IP 71707: Plant Operation
IP 71750: Plant Support Activities
IP 92700: LER Followup
IP 92903: Engineering Followup