April 7, 1995

Mr. D. N. Morey, Vice President
Southern Nuclear Operating Co., Inc.
Post Office Box 1295
Birmingham, AL 35201-1295

SUBJECT: ISSUANCE OF AMENDMENT NO. 114 TO FACILITY OPERATING LICENSE NO. NPF-2 AND AMENDMENT NO. 105 TO FACILITY OPERATING LICENSE NO. NPF-8 REGARDING RELOCATION OF THE LOWER LEVEL STEAM GENERATOR WATER LEVEL TAP - JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. M89839 AND M89840)

Dear Mr. Morey:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 114 to Facility Operating License No. NPF-2 and Amendment No. 105 to Facility Operating License No. NPF-8 for the Joseph M. Farley Nuclear Plant, Units 1 and 2. The amendments change the Technical Specifications (TS) in response to your submittal dated June 10, 1994.

The amendments allow modifications to be made for both units to relocate the lower level steam generator water level taps during the upcoming refueling outages. These modifications affect the TS associated with the reactor trip system and engineered safety feature actuation system setpoints.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

Original signed by:

Byron L. Siegel, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-348
and 50-364

DISTRIBUTION
See next page

Enclosures:
1. Amendment No. 114 to NPF-2
2. Amendment No. 105 to NPF-8
3. Safety Evaluation

cc w/enclosures:
See next page
*See previous concurrence
DOCUMENT NAME: G:\FARLEY\FAR89839.AMD

OFFICE	LA:PDII-2	PM:PDII-2	D:PDII-2	*OGC	
NAME	LBerry	BSiegel	HBerkow	RBachmann	
DATE	04/07/95	04/07/95	04/7/95	03/03/95	
COPY	Yes/No	Yes/No	Yes/No	Yes/No	

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 7, 1995

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Southern Nuclear Operating Co., Inc.
Post Office Box 1295
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SUBJECT: ISSUANCE OF AMENDMENT NO. 114 TO FACILITY OPERATING LICENSE NO. NPF-2 AND AMENDMENT NO. 105 TO FACILITY OPERATING LICENSE NO. NPF-8 REGARDING RELOCATION OF THE LOWER LEVEL STEAM GENERATOR WATER LEVEL TAP - JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. M89839 AND M89840)

Dear Mr. Morey:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 114 to Facility Operating License No. NPF-2 and Amendment No. 105 to Facility Operating License No. NPF-8 for the Joseph M. Farley Nuclear Plant, Units 1 and 2. The amendments change the Technical Specifications (TS) in response to your submittal dated June 10, 1994.

The amendments allow modifications to be made for both units to relocate the lower level steam generator water level taps during the upcoming refueling outages. These modifications affect the TS associated with the reactor trip system and engineered safety feature actuation system setpoints.

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Sincerely,

Byron L. Siegel

Byron L. Siegel, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-348 and 50-364

Enclosures:
1. Amendment No. 114 to NPF-2
2. Amendment No. 105 to NPF-8
3. Safety Evaluation

cc w/enclosures:
See next page

Mr. D. N. Morey
Southern Nuclear Operating
Company, Inc.

Joseph M. Farley Nuclear Plant

cc:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

DOCKET NO. 50-348

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 114
License No. NPF-2

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Southern Nuclear Operating Company, Inc. (Southern Nuclear), dated June 10, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. NPF-2 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 114 , are hereby incorporated in the license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented prior to restart from the next refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: April 7, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 114

TO FACILITY OPERATING LICENSE NO. NPF-2

DOCKET NO. 50-348

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Remove Pages	Insert Pages
2-6	2-6
3/4 3-27	3/4 3-27
3/4 3-28	3/4 3-28

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level--Low-Low	≥ 25% of narrow range instrument span - each steam generator	≥ 23.3% of narrow range instrument span - each steam generator
14. Deleted	--------	--------
15. Undervoltage - Reactor Coolant Pumps	≥ 2680 volts - each bus	≥ 2640 volts - each bus
16. Underfrequency Reactor Coolant Pumps	≥ 57.0 Hz - each bus	≥ 56.9 Hz - each bus
17. Turbine Trip		
A. Low Auto Stop Pressure	≥ 45 psig	≥ 43 psig
B. Turbine Stop Valve Closure	Not Applicable	Not Applicable
18. Safety Injection Input from ESF	Not Applicable	Not Applicable
19. Reactor Coolant Pump Breaker Position Trip	Not Applicable	Not Applicable

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
4. STEAM LINE ISOLATION		
a. Manual	Not Applicable	Not applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High-High	≤ 16.2 psig	≤ 17.5 psig
d. Steam Flow in Two Steam Lines--High, Coincident with Tavg--Low-Low	≤ A function defined as follows: A Δp corresponding to 40% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 110% of full steam flow at full load with T_{avg} ≥ 543°F	≤ A function defined defined as follows: A Δp corresponding to 44% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 111.5% of full steam flow at full load with T_{avg} ≥ 540°F
e. Steam Line Pressure--Low	≥ 585 psig	≥ 575 psig
5. TURBINE TRIP AND FEED WATER ISOLATION		
a. Steam Generator Water Level--High-High	≤ 79.2% of narrow range instrument span each steam generator	≤ 80.5% of narrow range instrument span each steam generator

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
6. AUXILIARY FEEDWATER		
a. Automatic Actuation Logic	N.A.	N.A.
b. Steam Generator Water Level--Low-Low	≥ 25% of narrow range instrument span each steam generator	≥ 23.3% of narrow range instrument span each steam generator
c. Undervoltage - RCP	≥ 2680 volts	≥ 2640 volts
d. S.I.	See 1 above (all SI setpoints)	
e. Trip of Main Feedwater Pumps	N.A.	N.A.
7. LOSS OF POWER		
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	≥ 3255 volts bus voltage*	≥ 3222 volts bus voltage* ≤ 3418 volts bus voltage*
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	≥ 3675 volts bus voltage*	≥ 3638 volts bus voltage* ≤ 3749 volts bus voltage*
8. ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INTERLOCKS		
a. Pressurizer Pressure, P-11	≤ 2000 psig	≤ 2010 psig
b. Low-Low T_{avg}, P-12		
(Increasing)	544°F	≤ 547°F
(Decreasing)	543°F	≥ 540°F
c. Steam Generator Level, P-14 (See 5. above)		
d. Reactor Trip, P-4	N.A.	N.A.

* Refer to appropriate relay setting sheet calibration requirements.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

DOCKET NO. 50-364

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 105
License No. NPF-8

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Southern Nuclear Operating Company, Inc. (Southern Nuclear), dated June 10, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. NPF-8 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 105, are hereby incorporated in the license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 7, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 105

TO FACILITY OPERATING LICENSE NO. NPF-8

DOCKET NO. 50-364

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Remove Pages	Insert Pages
2-6	2-6
3/4 3-27	3/4 3-27
3/4 3-28	3/4 3-28

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level--Low-Low	≥ 25% of narrow range instrument span - each steam generator	≥ 23.3% of narrow range instrument span - each steam generator
14. Deleted	--------	--------
15. Undervoltage - Reactor Coolant Pumps	≥ 2680 volts - each bus	≥ 2640 volts - each bus
16. Underfrequency Reactor Coolant Pumps	≥ 57.0 Hz - each bus	≥ 56.9 Hz - each bus
17. Turbine Trip		
A. Low Auto Stop Pressure	≥ 45 psig	≥ 43 psig
B. Turbine Stop Valve Closure	Not Applicable	Not Applicable
18. Safety Injection Input from ESF	Not Applicable	Not Applicable
19. Reactor Coolant Pump Breaker Position Trip	Not Applicable	Not Applicable

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
4. STEAM LINE ISOLATION		
a. Manual	Not Applicable	Not applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High-High	≤ 16.2 psig	≤ 17.5 psig
d. Steam Flow in Two Steam Lines--High, Coincident with Tavg--Low-Low	≤ A function defined as follows: A Δp corresponding to 40% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 110% of full steam flow at full load with T_{avg} ≥ 543°F	≤ A function defined defined as follows: A Δp corresponding to 44% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 111.5% of full steam flow at full load with T_{avg} ≥ 540°F
e. Steam Line Pressure--Low	≥ 585 psig	≥ 575 psig
5. TURBINE TRIP AND FEED WATER ISOLATION		
a. Steam Generator Water Level--High-High	≤ 79.2% of narrow range instrument span each steam generator	≤ 80.5% of narrow range instrument span each steam generator

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
6. AUXILIARY FEEDWATER		
a. Automatic Actuation Logic	N.A.	N.A.
b. Steam Generator Water Level--Low-Low	≥ 25% of narrow range instrument span each steam generator	≥ 23.3% of narrow range instrument span each steam generator
c. Undervoltage - RCP	≥ 2680 volts	≥ 2640 volts
d. S.I.	See 1 above (all SI setpoints)	
e. Trip of Main Feedwater Pumps	N.A.	N.A.
7. LOSS OF POWER		
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	≥ 3255 volts bus voltage*	≥ 3222 volts bus voltage* ≤ 3418 volts bus voltage*
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	≥ 3675 volts bus voltage*	≥ 3638 volts bus voltage* ≤ 3749 volts bus voltage*
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b. Low-Low T_{avg}, P-12		
(Increasing)	544°F	≤ 547°F
(Decreasing)	543°F	≥ 540°F
c. Steam Generator Level, P-14	(See 5. above)	
d. Reactor Trip, P-4	N.A.	N.A.

* Refer to appropriate relay setting sheet calibration requirements.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 114 TO FACILITY OPERATING LICENSE NO. NPF-2

AND AMENDMENT NO. 105 TO FACILITY OPERATING LICENSE NO. NPF-8

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-348 AND 50-364

1.0 INTRODUCTION

By letter dated June 10, 1994 (Reference 1), the Southern Nuclear Operating Company (SNC or the licensee) submitted proposed changes to the Technical Specification (TS) for the Joseph M. Farley Nuclear Plant (Farley) Units 1 and 2, to allow modifications to relocate the lower level steam generator water level taps to be made during the upcoming refueling outages to both units. These modifications affect the TS associated with the Reactor Trip System (RTS) and Engineered Safety Feature Actuation System (ESFAS) setpoints.

In support of these amendments, the licensee submitted, as an enclosure to the June 10, 1994, letter, a proprietary report prepared by Westinghouse, WCAP-13992 (Reference 2). A non-proprietary version of this report, WCAP-13993, was also submitted.

The following TS changes are proposed for Farley, Units 1 and 2:

(1) In Table 2.2-1, Reactor Trip System Instrumentation Trip Setpoints, Functional Unit 13 (Steam Generator Water Level Low-Low), the trip setpoint was changed from ≥ 15% to ≥ 25% of the narrow range instrument span and the allowable value was changed from ≥ 14.4% to ≥ 23.3% of the narrow range instrument span of each steam generator.

(2) In Table 3.3-4, Engineered Safety Feature Actuation System Instrumentation Trip Setpoints, for Functional Unit 5.a (Steam Generator Water Level High-High) the turbine trip and feed water isolation trip setpoint was changed from ≤ 75% to ≤ 79.2% of the narrow range instrument span and the allowable value was changed from ≤ 76% to ≤ 80.5% of the narrow range instrument span of each steam generator.

(3) For Functional Unit 6.b,in Table 3.3-4 (Steam Generator Water Level Low-Low) the trip actuation setpoint was changed from ≥ 15% to ≥ 25% of the narrow range instrument span and the allowable value was changed from ≥ 14.4% to ≥ 23.3% of the narrow range instrument span of each steam generator.

2.0 BACKGROUND

Both Farley units currently has reactor trip and safeguards actuation on low-low steam generator water level and reactor trip on steam/feedwater mismatch and low steam generator water level for protection for loss of heat sink caused by postulated events such as loss of normal feedwater, feedline rupture, and loss of all ac power to station auxiliaries. The steam generator low-low water level trip setpoint change reflects the proposed Farley modification to the lower level taps, including Farley specific instrumentation, procedures, calibration practices and uncertainties, and accounts for the increased span due to lowering of the steam generator lower level taps. In addition, the steam generator high-high level setpoint for turbine trip and feedwater isolation has been revised to be consistent with the increased narrow range span. Reference 2 provides the basis of the revised setpoints. The purpose of the revision to these setpoints is to allow increased operational flexibility and to reduce spurious reactor trips due to feedwater system transients.

3.0 EVALUATION

Because the relocation of the level taps results in a reduced steam generator inventory, the applicable transients and accidents related to the Farley design basis required review and/or revision. SNC stated that all non-LOCA analyses that credit low-low steam generator level as primary protection were re-analyzed. In addition, the most limiting steamline breaks for environmental qualification which credit low-low steam generator level as primary protection were re-analyzed. Setpoint uncertainty calculations were also performed for steam generator level low-low level reactor trip and ESFAS high-high level turbine trip and feedwater isolation. The instrumentation uncertainties associated with each of these protective system functions were calculated using the Westinghouse statistical setpoint methodology.

3.1 Transient and Accident Safety Evaluations

3.1.1 Non-LOCA Evaluation

The steam generator level tap relocation and the low-low level setpoint reduction required that the following Final Safety Analysis Report (FSAR) Chapter 15 accident analyses be re-analyzed: (1) the loss of normal feedwater (FSAR Section 15.2.8), (2) loss of non-emergency ac power to plant auxiliaries (FSAR Section 15.2.9), and (3) the feedwater system pipe break (FSAR Section 15.4.2.2). These were re-analyzed by SNC as discussed below.

3.1.1.1 Loss of Normal Feedwater and Loss of Non-Emergency AC Power to the Plant Auxiliaries

The loss of normal feedwater and the loss of non-emergency ac power to the plant auxiliaries are American Nuclear Society (ANS) Condition II events that are analyzed to demonstrate adequate heat removal capability exists to remove core decay heat and stored energy following reactor trip. The acceptance criteria for these events include demonstrating there is no overpressurization of the primary or secondary side and that pressurizer filling does not occur.

The physical relocation of the steam generator level tap and the corresponding reduction in the low-low level setpoint reduces the amount of mass available following reactor trip to remove the core decay heat and stored energy, resulting in a potentially more limiting transient.

The licensee stated that the analysis method and analysis assumptions were the same as used in the current Farley FSAR. SNC stated that the results of the transient analyses showed that the capacity of the auxiliary feedwater system is adequate to provide sufficient heat removal from the Reactor Coolant System (RCS) following a reactor trip. The criterion that the pressurizer does not fill was met, assuring that the integrity of the primary system is not adversely affected. For the case without offsite power available, the results verified that the natural circulation capacity of the RCS provides sufficient heat removal capability to prevent fuel or clad damage following reactor coolant pump coastdown.

3.1.1.2 Feedwater System Pipe Break

The feedwater system pipe break is an ANS Condition IV event which is analyzed to demonstrate that the peak primary and secondary side pressures do not exceed allowable limits and the core remains adequately covered with water. The relocation of the steam generator low-low level setpoint reduces the amount of mass available following a reactor trip to remove the core decay heat and stored energy, resulting in a potentially more limiting transient.

The licensee stated that two cases were analyzed in the Farley FSAR which vary the auxiliary feedwater (AFW) delivered to the intact steam generators following actuation on a steam generator low-low level signal. The first (Case A) assumes a total AFW flow rate of 350 gpm from the two motor driven pumps delivered to two steam generators 10 minutes following an actuation signal on low-low level. The second (Case B) assumes a total 150 gpm is fed to the intact steam generators on a low-low level signal following a 60-second delay. The flow is then increased to 350 gpm 30 minutes from the time of the actuation signal. Additional key assumptions are: (1) the initial power is assumed to be at 102% of the NSSS design power rating (2,790 MWt); (2) a conservative core residual heat generation model based on the 1979 version of ANS-5.1 is used; (3) the steam generator low-low water level setpoint is conservatively assumed to be at 0% of the new narrow range span; and (4) a 20% steam generator tube plugging level is also assumed. The method of analysis and assumptions used are otherwise in accordance with those presented in the FSAR.

The licensee stated that the transient results show that the capacity of the auxiliary feedwater system is adequate to provide sufficient heat removal from the RCS to prevent overpressurization of the RCS and the main steam system and to prevent core uncovery. The reactor coolant remains subcooled, assuring that the core remains adequately covered with water. The analysis results also verify that the natural circulation capacity of the RCS provides sufficient heat removal capability following reactor coolant pump coastdown.

3.1.1.3 Non-LOCA Transients Not Requiring Any Reanalysis

The licensee stated that the following transients were not reanalyzed since either the transients are not affected by safety analysis assumptions or any change to secondary side analysis assumptions will not adversely affect the results of the analyses.

Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition (FSAR Section 15.2.1)

Uncontrolled RCCA Bank Withdrawal at Power (FSAR Section 15.2.2)

RCCA Misalignment (FSAR Section 15.2.3)

Uncontrolled Boron Dilution (FSAR Section 15.2.4)

Partial Loss of Forced Reactor Coolant Flow (FSAR Section 15.2.5)

Startup of an Inactive Reactor Coolant Loop (FSAR Section 15.2.6)

Loss of External Electrical Load and/or Turbine Trip (FSAR Section 15.2.7)

Excessive Heat Removal Due to Feedwater System Malfunctions (FSAR Section 15.2.10)

Excessive Load Increase Incident (FSAR Section 15.2.11)

Accidental Depressurization of the RCS (FSAR Section 15.2.12)

Accidental Depressurization of the Main Steam System (FSAR Section 15.2,13)

Inadvertent Operation of ECCS During Power Operation (FSAR Section 15.2.14)

Complete Loss of Forced Reactor Coolant Flow (FSAR Section 15.3.4)

Rupture of Main Steam Line (FSAR Section 15.4.2.1)

Single Reactor Coolant Pump Locked Rotor (FSAR Section 15.4.4)

Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection) (FSAR Section 15.4.6)

Main Steam Line Ruptures Inside Containment (FSAR Section 6.2.1.3.11)

In addition, SNC stated that all non-LOCA analysis, including main steam line break mass and energy releases for containment response, which were not specifically reanalyzed, were evaluated and found not to be impacted by the proposed modification.

3.1.2 LOCA Related Evaluations

The licensee stated that all LOCA, LOCA forces, steam generator tube rupture, and LOCA-related analyses were reviewed and were found to be unaffected by this proposed modification.

3.1.3 Steam Generator Water Level Control

The licensee stated that the steam generator water level control system uses inputs from narrow range level instruments. Therefore, the control system programmed setpoint will be revised to account for the increased fluid velocity effect and the increased span resulting from the relocated lower level tap.

3.2 Setpoint Methodology Evaluations

The Farley Units 1 and 2 steam generator level instrumentation lower tap modification moved the tap approximately 68 inches (from 443 inches to 375 inches). This relocation resulted in an increased level span from 144 inches to 212 inches. The functions affected by the change are; steam generator low-low level reactor trip and ESF actuation and the steam generator high-high level turbine trip and feedwater isolation. The instrument uncertainties associated with each of these protection system functions were calculated using the Westinghouse statistical setpoint methodology. The calculations accounted for all known instrument uncertainties associated with the level transmitters, signal processing equipment, and calibration methods that are applicable to these functions. In addition, process measurement accuracy allowances were included to account for process pressure changes and reference leg ambient temperature changes from the reference conditions, as well as fluid velocity effects and downcomer subcooling effects associated with the new lower tap location. Environmental allowances were also included in the level setpoint calculations to account for the potential effects induced on the level transmitter, signal cable and reference leg by adverse containment environmental conditions. This calculation resulted in the proposed Nominal Trip Setpoints of 25% narrow range span (NRS) for low-low level and 79.2% NRS for high-high level which provides positive margin to the Safety Analysis Limits, after accounting for all known uncertainties. The allowable values for the low-low and high-high steam generator level protection functions have been calculated to be 23.3% NRS and 80.5% NRS, respectively. Comparison of the existing setpoint (15% NRS for low-low and 75% NRS for high-high level trip) with the proposed values indicate that the proposed setpoints are conservative, will allow increased operational flexibility, and will reduce spurious reactor trips due to feedwater system transients.

The licensee further committed to revise the Control System programmed setpoint to account for the fluid velocity effect and the increased narrow range span.

The staff compared SNC's calculations with those contained in WCAP-13751 (Reference 3) which were previously reviewed and approved by the staff.

3.3 Evaluation Summary

The staff has reviewed the safety analyses and setpoint evaluations performed by SNC to support the proposed level tap modification and subsequent setpoint changes. Since the results of the reanalyzed accidents are within allowable limits and the proposed setpoint methodology calculations are consistent with those contained in WCAP-13751, which were previously approved by the staff, the staff concludes that the proposed TS changes resulting from the modification to lower the steam generator level taps are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of Alabama official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (60 FR 12253 dated March 6, 1995). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: H. Balukjian
I. Ahmed

Date: April 7, 1995

REFERENCES

1. Letter from D. Morey, SNC, to USNRC, June 10, 1994.

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AMENDMENT NO. 114 TO FACILITY OPERATING LICENSE NO. NPF-2 - FARLEY, UNIT 1
AMENDMENT NO. 105 TO FACILITY OPERATING LICENSE NO. NPF-8 - FARLEY, UNIT 2

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