December 29, 1993

Docket Nos. 50-348
and 50-364

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

Dear Mr. Morey:

SUBJECT: ISSUANCE OF AMENDMENT NO. 104 TO FACILITY OPERATING LICENSE NO. NPF-2 AND AMENDMENT NO. 97 TO FACILITY OPERATING LICENSE NO. NPF-8 REGARDING LOW FEEDWATER FLOW REACTOR TRIP ELIMINATION AND LOW-LOW STEAM GENERATOR LEVEL SETPOINT CHANGE - JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. M87700 AND M87701)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 104 to Facility Operating License No. NPF-2 and Amendment No. 97 to Facility Operating License No. NPF-8 for the Joseph M. Farley Nuclear Plant, Units 1 and 2. The amendments change the Technical Specifications in response to your submittal dated September 13, 1993.

The amendments change the Technical Specifications to eliminate the low feedwater reactor trip and reduce the steam generator low-low water level reactor trip and safeguard actuation setpoint from 17 percent to 15 percent of narrow range span with a corresponding reduction in allowable value from 16 percent to 14.4 percent.

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-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 104 to NPF-2
2. Amendment No. 97 to NPF-8
3. Safety Evaluation

cc w/enclosures:
See next page

CP-1

*See Previous Concurrence

OFC	LA:PDII-1	PM:PDII-1	D:PDII-1	OGC*	
NAME	PAnderson	BSiegel:jrm	SBajwa	RBachmann	
DATE	11/29/93	11/29/93	11/29/93	11/18/93	

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

December 29, 1993

Docket Nos. 50-348
and 50-364

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Post Office Box 1295
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Dear Mr. Morey:

SUBJECT: ISSUANCE OF AMENDMENT NO. 104 TO FACILITY OPERATING LICENSE NO. NPF-2 AND AMENDMENT NO. 97 TO FACILITY OPERATING LICENSE NO. NPF-8 REGARDING LOW FEEDWATER FLOW REACTOR TRIP ELIMINATION AND LOW-LOW STEAM GENERATOR LEVEL SETPOINT CHANGE - JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. M87700 AND M87701)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 104 to Facility Operating License No. NPF-2 and Amendment No. 97 to Facility Operating License No. NPF-8 for the Joseph M. Farley Nuclear Plant, Units 1 and 2. The amendments change the Technical Specifications in response to your submittal dated September 13, 1993.

The amendments change the Technical Specifications to eliminate the low feedwater reactor trip and reduce the steam generator low-low water level reactor trip and safeguard actuation setpoint from 17 percent to 15 percent of narrow range span with a corresponding reduction in allowable value from 16 percent to 14.4 percent.

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,

Byron L. Siegel

Byron L. Siegel, Senior Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 104 to NPF-2
2. Amendment No. 97 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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Columbia, Alabama 36319

AMENDMENT NO. 104 TO FACILITY OPERATING LICENSE NO. NPF-2 - FARLEY, UNIT 1
AMENDMENT NO. 97 TO FACILITY OPERATING LICENSE NO. NPF-8 - FARLEY, UNIT 2

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030098

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. 104
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 September 13, 1993, 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. 104 , 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 startup from refueling outage 12.

FOR THE NUCLEAR REGULATORY COMMISSION

Bajwa

S. Singh Bajwa, Acting Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 29, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 104

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
B 2-6	B 2-6
3/4 3-3	3/4 3-3
3/4 3-11	3/4 3-11
3/4 3-13	3/4 3-13
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	15% of narrow range instrument span - each steam generator	≥ 14.4% 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

latter trip will ensure that the DNB design criterion is met during normal operational transients and anticipated transients when 2 loops are in operation and the Overtemperature delta T trip setpoint is adjusted to the value specified for all loops in operation. With the Overtemperature delta T trip setpoint adjusted to the value specified for 2 loop operation, the P-8 trip at 66% RATED THERMAL POWER will ensure that the DNB design criterion is met during normal operational transients and anticipated transients with 2 loops in operation.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity. The specified setpoint provides allowances that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays of the auxiliary feedwater system.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump bus trips provide reactor core protection against DNB as a result of loss of voltage or underfrequency to more than one reactor coolant pump. The specified setpoints assure a reactor trip signal is generated before the low flow trip setpoint

TABLE 3.3-1 (CONTINUED)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. Pressurizer Water Level--High	3	2	2	1	7#
12. A. Loss of Flow - Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	7#
B. Loss of Flow - Two Loops (Above P-7 and Below P-8)	3/loop	2/loop in two operating loops	2/loop in each operating loop	1	7#
13. Steam Generator Water Level--Low-Low	3/loop	2/loop in any operating loops	2/loop in each operating loop	1,2	7#
14. Deleted	———	———	———	———	———
15. Undervoltage - Reactor Coolant Pumps	3-2/bus	2	2	1	7#
16. Underfrequency - Reactor Coolant Pumps	3-2/bus	2	2	1	7#

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
12. A. Loss of Flow - Single Loop (Above P-8)	≤ 1.0 seconds
B. Loss of Flow - Two Loops (Above P-7 and Below P-8)	≤ 1.0 seconds
13. Steam Generator Water Level--Low-Low	≤ 2.0 seconds
14. Deleted	———
15. Undervoltage - Reactor Coolant Pumps	≤ 1.2 seconds
16. Underfrequency - Reactor Coolant Pumps	≤ 0.6 seconds
17. Turbine Trip	
A. Low Auto Stop Oil Pressure	Not Applicable
B. Turbine Throttle Valve Closure	Not Applicable
18. Safety Injection Input from ESF	Not Applicable
19. Reactor Coolant Pump Breaker Position Trip	Not Applicable
20. Reactor Trip System Interlocks	Not Applicable
21. Reactor Trip Breakers	Not Applicable
22. Automatic Trip Logic	Not Applicable

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
13. Steam Generator Water Level--Low-Low	S	R	M	1, 2
14. Deleted	———	———	———	———
15. Undervoltage - Reactor Coolant Pumps	N.A.	R	M	1
16. Underfrequency - Reactor Coolant Pumps	N.A.	R	M	1
17. Turbine Trip				
A. Low Auto Stop Oil Pressure	N.A.	N.A.	S/U(9)(1)	N.A.
B. Turbine Throttle Valve Closure	N.A.	N.A.	S/U(9)(1)	N.A.
18. Safety Injection Input from ESF	N.A.	N.A.	M(4)	1, 2
19. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	R	1
20. Reactor Trip System Interlocks	N.A.	R	S/U(8)	1
21. Reactor Trip Breaker	N.A.	N.A.	M(5)(14)(15), S/U(1)(14)(15)	1, 2, 3*, 4*, 5*
22. Automatic Trip Logic	N.A.	N.A.	M(5)	1, 2, 3*, 4*, 5*
23. Reactor Trip Bypass Breaker	N.A.	N.A.	(13),R(11)	1, 2, 3*, 4*, 5*

TABLE 3.3-4 (Continued)

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	≥ 15% of narrow range instrument span each steam generator	≥ 14.4% 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. 97
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 September 13, 1993, 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. 97 , 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 startup from refueling outage 10.

FOR THE NUCLEAR REGULATORY COMMISSION

Bajwa

S. Singh Bajwa, Acting Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 29, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 97

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
B 2-6	B 2-6
3/4 3-3	3/4 3-3
3/4 3-11	3/4 3-11
3/4 3-13	3/4 3-13
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	15% of narrow range instrument span - each steam generator	≥ 14.4% 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

latter trip will ensure that the DNB design criterion is met during normal operational transients and anticipated transients when 2 loops are in operation and the Overtemperature delta T trip setpoint is adjusted to the value specified for all loops in operation. With the Overtemperature delta T trip setpoint adjusted to the value specified for 2 loop operation, the P-8 trip at 66% RATED THERMAL POWER will ensure that the DNB design criterion is met during normal operational transients and anticipated transients with 2 loops in operation.

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Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump bus trips provide reactor core protection against DNB as a result of loss of voltage or underfrequency to more than one reactor coolant pump. The specified setpoints assure a reactor trip signal is generated before the low flow trip setpoint

TABLE 3.3-1 (CONTINUED)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. Pressurizer Water Level--High	3	2	2	1	7#
12. A. Loss of Flow - Single Loop (Above P-8)	3/loop	2/loop in any oper- ating loop	2/loop in each oper- ating loop	1	7#
B. Loss of Flow - Two Loops (Above P-7 and Below P-8)	3/loop	2/loop in two oper- ating loops	2/loop in each oper- ating loop	1	7#
13. Steam Generator Water Level-- Low-Low	3/loop	2/loop in any oper- ating loops	2/loop in each oper- ating loop	1,2	7#
14. Deleted	———	———	———	———	———
15. Undervoltage - Reactor Coolant Pumps	3-2/bus	2	2	1	7#
16. Underfrequency - Reactor Coolant Pumps	3-2/bus	2	2	1	7#

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
12. A. Loss of Flow - Single Loop (Above P-8)	≤ 1.0 seconds
B. Loss of Flow - Two Loops (Above P-7 and Below P-8)	≤ 1.0 seconds
13. Steam Generator Water Level--Low-Low	≤ 2.0 seconds
14. Deleted	———
15. Undervoltage - Reactor Coolant Pumps	≤ 1.2 seconds
16. Underfrequency - Reactor Coolant Pumps	≤ 0.6 seconds
17. Turbine Trip	
A. Low Auto Stop Oil Pressure	Not Applicable
B. Turbine Throttle Valve Closure	Not Applicable
18. Safety Injection Input from ESF	Not Applicable
19. Reactor Coolant Pump Breaker Position Trip	Not Applicable
20. Reactor Trip System Interlocks	Not Applicable
21. Reactor Trip Breakers	Not Applicable
22. Automatic Trip Logic	Not Applicable

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
13. Steam Generator Water Level--Low-Low	S	R	M	1, 2
14. Deleted	———	———	———	———
15. Undervoltage - Reactor Coolant Pumps	N.A.	R	M	1
16. Underfrequency - Reactor Coolant Pumps	N.A.	R	M	1
17. Turbine Trip				
A. Low Auto Stop Oil Pressure	N.A.	N.A.	S/U(9)(1)	N.A.
B. Turbine Throttle Valve Closure	N.A.	N.A.	S/U(9)(1)	N.A.
18. Safety Injection Input from ESF	N.A.	N.A.	M(4)	1, 2
19. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	R	1
20. Reactor Trip System Interlocks	N.A.	R	S/U(8)	1
21. Reactor Trip Breaker	N.A.	N.A.	M(5)(14)(15), S/U(1)(14)(15)	1, 2, 3*, 4*, 5*
22. Automatic Trip Logic	N.A.	N.A.	M(5)	1, 2, 3*, 4*, 5*
23. Reactor Trip Bypass Breaker	N.A.	N.A.	(13),R(11)	1, 2, 3*, 4*, 5*

TABLE 3.3-4 (Continued)

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	≥ 15% of narrow range instrument span each steam generator	≥ 14.4% 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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 104 TO FACILITY OPERATING LICENSE NO. NPF-2

AND AMENDMENT NO. 97 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 September 13, 1993, Southern Nuclear Operating Company, Inc. (the licensee), submitted a request for changes to the Joseph M. Farley Nuclear Plant, Units 1 and 2 (Farley), Technical Specifications (TS). The proposed changes eliminate the low feedwater flow reactor trip (i.e., steam flow/feedwater flow mismatch in coincidence with low steam generator level) from the TS following the installation of a median signal selector (MSS) in the steam generator water level control system. The proposed amendment also reduces the low-low steam generator level trip setpoint and allowable value. Specifically, the TS changes for both units include the following:

(1) The Trip Setpoint and Allowable Value in Item 13 of TS Table 2.2-1, Reactor Trip System Instrumentation Trip Setpoints, would be reduced from 17 percent to 15 percent and from 16 percent to 14.1 percent, respectively.

(2) Item 14 of TS Table 2.2-1, Reactor Trip System Instrumentation Trip Setpoints, and the associated Bases would be deleted.

(3) Item 14 of TS Table 3.3-1, Reactor Trip System Instrumentation, would be deleted.

(4) Item 14 of TS Table 3.3-2, Reactor Trip System Instrumentation Response Times, would be deleted.

(5) The trip setpoint and allowable value in Item 6.b. of TS Table 3.3-4, Engineered Safety Feature Actuation System Instrumentation Trip Setpoints, would be reduced from 17 percent to 15 percent and from 16 percent to 14.4 percent, respectively.

(6) Item 14 of TS Table 4.3-1, Reactor Trip System Instrumentation Surveillance Requirements, would be deleted.

In support of these amendments the licensee submitted, as enclosures to the September 13, 1993, letter, two proprietary reports prepared by Westinghouse Electric Corporation: WCAP-13807, "Elimination of the Low Feedwater Flow Reactor Trip via Implementation of the Median Signal Selector (MSS) at Farley Units 1 and 2," and WCAP-13751, "Westinghouse Setpoint Methodology for Protection Systems - Farley Nuclear Plant Units 1 and 2."

2.0 EVALUATION

Each of the steam generators at Farley, Units 1 and 2, has three independent narrow range water level detection instrument channels which provide input to the reactor protection system (RPS). The low-low steam generator water level protection function is configured with two-out-of-three actuation logic derived directly from these three narrow range level channels for each steam generator. The two-out-of-three actuation logic also provides a starting signal for the auxiliary feedwater pumps. The low-low steam generator water level reactor trip function is designed to preserve the steam generator as a heat sink for removal of residual heat if there is a loss of normal feedwater. The low feedwater flow reactor trip is configured to initiate a reactor trip during a condition of steam and feedwater flow mismatch on one-out-of-two channels in coincidence with low steam generator water level on one-out-of-two channels. The Farley FSAR does not assume that the low feedwater flow reactor trip mitigates the consequences of any analyzed accident. In events such as a loss of normal feedwater or loss of all AC power, credit is only taken for the low-low steam generator water level reactor trip to ensure safe shutdown of the reactor.

At Farley, one of the steam generator water level instrument channels also supplies an input to the steam generator water level control system. As a result, a common instrument channel is used for both the RPS and the steam generator water level control system, separated electrically with a qualified isolation device. The low feedwater flow reactor trip was installed to satisfy the requirements of the Institute of Electric and Electronics Engineers Standard 279, 1971 (IEEE Std. 279), "Criterion for Protection Systems for Nuclear Power Generating Stations," which is endorsed by 10 CFR Part 50.55a. Section 4.7.3, "Single Random Failure," of IEEE Std. 279 states, in part, "where a single random failure can cause a control system action that results in a generating station condition requiring protective action and also prevent proper action of a protective system channel designed to protect against the condition, the remaining redundant protection channels shall be capable of providing the protective action even when degraded by a second random failure." The purpose of the low feedwater reactor trip was to satisfy this criterion.

The staff reviewed WCAP-13807, "Elimination of the Low Feedwater Flow Reactor Trip via Implementation of the Median Signal Selector at Farley Units 1 and 2," which provides a detailed description regarding the addition of the median signal selector (MSS) to the steam generator water level control system. The MSS is a hardware device that is designed to select the median of the three narrow range steam generator water level instrument input signals. By selecting the median signal, a single random failure will not cause a control system action that results in a condition requiring protective action. This is because the failure of a single protection channel will not result in adverse control system behavior because the median signal of the three level channels will be selected for control purposes. Thus, the possibility of adverse interaction between the steam generator water level control system and the RPS due to a single failure is eliminated, thereby satisfying the requirements of IEEE Std. 279 without credit being taken for the low feedwater flow reactor trip.

The steam generator water level control system MSS is functionally the same as the MSS design currently being used for the average temperature and delta-temperature control circuits in the RTD bypass elimination modification at Farley Unit 1. The MSS used in the RTD bypass elimination circuitry was approved by the staff as documented in Amendment No. 87 to the Farley operating license dated March 8, 1993.

All three outputs from the steam generator narrow-range level channels are processed in the Westinghouse 7300 Process Protection Racks of the RPS, and are then input to the respective MSS. The single card MSS consists of operational amplifiers configured with input auctioneering (low and high), feedback resistance and diode networks and adjustable input and output signal conditioning to produce an output signal to the steam generator water level control system that is equal (in voltage) to the median of the three narrow range steam generator level input signals. The MSS electronics are of a quality consistent with low failure rates and minimum maintenance requirements, and these components conform to protection system design requirements.

A separate relay card will be added to the steam generator water level control system to provide the capability for on-line testing of the MSS downstream of the protection system isolation devices and for calibration of the MSS. The relay card allows for testing of the MSS during plant operation without placing the low-low steam generator channels in trip. The Farley operating and maintenance procedures will be revised to ensure that the procedures are consistent with, and support operation with an MSS, including administrative controls for operations with the MSS disabled or in test mode. In addition, during testing of the steam generator narrow range level protection channels, the output of the MSS will be observed. MSS calibration and functional testing will be included in the steam generator water level control system instrumentation calibration procedure which is currently conducted on a refueling basis and administratively controlled by the plant preventive maintenance program.

The isolation devices separating the low-low steam generator water level protection channels and the MSS of the steam generator water level control system are standard Westinghouse 7300 series equipment and were previously reviewed under WCAP-8892-A, "Westinghouse 7300 Series Process Control System Noise Tests," and accepted by the staff.

The licensee has stated that alarms will be added to the steam generator water level control system upon installing the MSS to indicate steam flow and feedwater flow mismatches and low steam generator level. In addition, annunciators will be added to alert the operator in the event of a power supply failure on the MSS card and whenever the relay card test injection switches are enabled.

The licensee has also requested to revise the steam generator water level low-low reactor trip setpoint and the auxiliary feedwater actuation setpoint from 17 percent to 15 percent. The revised low-low steam generator water level reactor trip setpoint was calculated using standard Westinghouse methodology as presented in WCAP-13751, "Westinghouse Setpoint Methodology for Protection Systems - Farley Nuclear Plant Units 1 and 2." This methodology is consistent with the Instrument Society of America Standard S67.04, 1987, "Setpoints for Nuclear Safety-Related Instrumentation used in Nuclear Power Plants.

The required setpoint is obtained by adding the specific channel statistical allowance (for channel inaccuracies) to the safety analysis limit. The safety analysis limit for the low-low steam generator water level is 0 percent of narrow range level span. The channel statistical allowance was calculated to be 14.9 percent of the instrument span. Thus, the results of the setpoint calculation indicate that sufficient margin exists to support a setpoint reduction from 17 percent to 15 percent without changing the safety analysis limit. The revised setpoint of 15 percent still meets the safety analysis limit with the required channel accuracies included and a 0.1 percent additional margin. In addition, calculations indicate that the allowable value can be reduced from 16 percent to 14.4 percent. The licensee has stated that the revised trip setpoint would increase the margin between the steam generator water level low-low reactor trip and the normal operating band thereby providing additional margin to spurious reactor trips. Based on the above information, the staff finds the proposed changes acceptable.

3.0 CONCLUSION

In summary, a review of the Farley safety analysis shows that no credit is taken for the reactor trip initiated by the low feedwater flow reactor trip in mitigating the consequences of any of the analyzed design basis accidents or transients. This trip was originally designed to satisfy the single random failure requirement specified in IEEE Std. 279, Section 4.7.3, for preventing adverse control and protective systems interaction. The MSS provides an acceptable alternative method of preventing interaction between the control and protection functions. As discussed above, the staff finds that utilization of the MSS is acceptable. Thus, with the addition of the MSS, the Farley steam generator water level control system meets the requirements of Section 4.7.3 of IEEE Std. 279 without the low feedwater flow reactor trip. On this basis, the staff finds the proposed TS changes involving the elimination of the low feedwater flow reactor trip to be acceptable following the installation of the MSS.

The revised setpoints for the low-low steam generator water level reactor trip and auxiliary feedwater actuation are consistent with the safety limit assumed in the FSAR analysis and are consistent with approved setpoint methodology. In addition, the revised setpoints remove an over conservatism which contributes to unnecessary reactor trips. On this basis, the staff finds the proposed change in the low-low steam generator water level reactor trip setpoint and allowable value to be 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 amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes 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 amendment involves 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding (58 FR 62157). Accordingly, the amendment meets 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 amendment.

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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date: December 29, 1993