June 16, 1994

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. 108 TO FACILITY OPERATING LICENSE NO. NPF-2 AND AMENDMENT NO. 100 TO FACILITY OPERATING LICENSE NO. NPF-8 REGARDING OVERPRESSURE PROTECTION SYSTEMS - JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. M77419 AND M77420)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 108 to Facility Operating License No. NPF-2 and Amendment No. 100 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 May 13, 1991, as supplemented October 13, 1992.

The amendments modify the TS for the overpressure protection systems. The allowable outage time (AOT) for one inoperable residual heat removal (RHR) relief valve with one or more of the reactor coolant system cold leg temperatures less than or equal to 310 degrees Fahrenheit is being decreased from 7 days to 24 hours for water-solid conditions. The required AOT for low temperature conditions, other than water-solid, will remain at 7 days with one RHR relief valve inoperable, provided the pressurizer level is less than or equal to 30 percent and a dedicated operator is assigned to monitor and control the reactor coolant system pressure.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's biweekly <u>Federal Register</u> 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. 108 to NPF-2
2. Amendment No. 100 to NPF-8
3. Safety Evaluation

cc w/enclosures:
See next page

* See previous concurrence

OFC	LA:PDII-1	PM:PDII-1	RSB*	D:PDII-1	OGC*
NAME	PAnderson	BSiegel/rsl	TCollins	WBateman	CWoodhead
DATE	6/16/94	6/16/94	06/02/94	6/16/94	06/07/94

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

June 16, 1994

Docket Nos. 50-348
and 50-364

Mr. D. N. Morey, Vice President
Southern Nuclear Operating Co., Inc.
Post Office Box 1295
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The amendments modify the TS for the overpressure protection systems. The allowable outage time (AOT) for one inoperable residual heat removal (RHR) relief valve with one or more of the reactor coolant system cold leg temperatures less than or equal to 310 degrees Fahrenheit is being decreased from 7 days to 24 hours for water-solid conditions. The required AOT for low temperature conditions, other than water-solid, will remain at 7 days with one RHR relief valve inoperable, provided the pressurizer level is less than or equal to 30 percent and a dedicated operator is assigned to monitor and control the reactor coolant system pressure.

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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. 108 to NPF-2
2. Amendment No. 100 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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AMENDMENT NO. 108 TO FACILITY OPERATING LICENSE NO. NPF-2 - FARLEY, UNIT 1
AMENDMENT NO. 100 TO FACILITY OPERATING LICENSE NO. NPF-8 - FARLEY, UNIT 2

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cc: Farley Service List

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. 108
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 May 13, 1991, as supplemented October 13, 1992, 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. 108 , 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

William H. Bateman

William H. Bateman, 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: June 16, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 108

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
3/4 4-32	3/4 4-32
B 3/4 4-14	B 3/4 4-14
---	B 3/4 4-14a

LIMITING CONDITIONS FOR OPERATIONS

3.4.10.3 At least one of the following overpressure protection systems shall be OPERABLE:

a. Two RHR relief valves with:

   1. A lift setting of less than or equal to 450 psig, and

   2. The associated RHR relief valve isolation valves open; or

b. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2.85 square inches.

APPLICABILITY: When the temperature of one or more of the RCS cold legs is less than or equal to 310°F, except when the reactor vessel head is removed.

ACTION:

a. With one RHR relief valve inoperable, restore the inoperable valve to OPERABLE status within 24 hours or perform the following:

   1. Establish the following requirements:

      i. Reduce pressurizer level to less than or equal to 30 percent (cold calibrated), and

      ii. Assign a dedicated operator for RCS pressure monitoring and control, and

      iii. Restore the inoperable valve to OPERABLE status within 7 days, or;

   2. Depressurize and vent the RCS through a greater than or equal to 2.85 square inch vent within the next 8 hours.

b. With both RHR relief valves inoperable, within 8 hours either:

   1. Restore at least one RHR relief valve to OPERABLE status, or

   2. Depressurize and vent the RCS through a greater than or equal to 2.85 square inch vent.

c. In the event a RHR relief valve or a RCS vent is used to mitigate a RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the RHR relief valves or vent on the transient and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the 10 CFR Part 50, Appendix G Rule which addresses the metal temperature of the closure head flange and vessel flange must be considered. This Rule states that the minimum metal temperature of the closure flange regions be at least 120°F higher than the limiting RT_{ndt} for these regions when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (621 psig for Farley Unit 1). In addition, the new 10 CFR Part 50 Rule states that a plant specific fracture evaluation may be performed to justify less limiting requirements. As a result, such a fracture analysis was performed for Farley Unit 2. These Farley Unit 2 fracture analysis results are applicable to Farley Unit 1 since the pertinent parameters are identical for both plants. Based upon this fracture analysis, the 16 EFPY heatup and cooldown curves are impacted by the new 10 CFR Part 50 Rule as shown on Figures 3.4-2 and 3.4-3.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of either RHR relief valve or an RCS vent opening of greater than or equal to 2.85 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 310°F. Either RHR relief valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures provided measures are taken to cushion the overpressure effects at RCS temperatures above 250°F, or (2) the start of 3 charging pumps and their injection into a water solid RCS.

3/4.4.11 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a(g)(6)(i).

3/4.4.12 REACTOR VESSEL HEAD VENTS

The OPERABILITY of the Reactor Head Vent System ensures that adequate core cooling can be maintained in the event of the accumulation of non-condensable gases in the reactor vessel. This system is in accordance with 10 CFR 50.44(c)(3)(iii).

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. 100
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 May 13, 1991, as supplemented October 13, 1992, 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. 100, 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

William H. Bateman

William H. Bateman, 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: June 16, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 100

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
3/4 4-32	3/4 4-32
B 3/4 4-14	B 3/4 4-14
---	B 3/4 4-14a

3.4.10.3 At least one of the following overpressure protection systems shall be OPERABLE:

a. Two RHR relief valves with:

   1. A lift setting of less than or equal to 450 psig, and

   2. The associated RHR relief valve isolation valves open; or

b. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2.85 square inches.

APPLICABILITY: When the temperature of one or more of the RCS cold legs is less than or equal to 310°F, except when the reactor vessel head is removed.

ACTION:

a. With one RHR relief valve inoperable, restore the inoperable valve to OPERABLE status within 24 hours or perform the following:

   1. Establish the following requirements:

      i. Reduce pressurizer level to less than or equal to 30 percent (cold calibrated), and

      ii. Assign a dedicated operator for RCS pressure monitoring and control, and

      iii. Restore the inoperable valve to OPERABLE status within 7 days, or;

   2. Depressurize and vent the RCS through a greater than or equal to 2.85 square inch vent within the next 8 hours.

b. With both RHR relief valves inoperable, within 8 hours either:

   1. Restore at least one RHR relief valve to OPERABLE status, or

   2. Depressurize and vent the RCS through a greater than or equal to 2.85 square inch vent.

c. In the event a RHR relief valve or a RCS vent is used to mitigate a RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the RHR relief valves or vent on the transient and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the 10 CFR Part 50, Appendix G Rule which addresses the metal temperature of the closure head flange and vessel flange must be considered. This Rule states that the minimum metal temperature of the closure flange regions be at least 120°F higher than the limiting RT_{ndt} for these regions when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (621 psig for Farley Unit 2). In addition, the new 10 CFR Part 50 Rule states that a plant specific fracture evaluation may be performed to justify less limiting requirements. Based upon such a fracture analysis for Farley Unit 2, the 14 EFPY heatup and cooldown curves are impacted by the new 10 CFR Part 50 Rule as shown on Figures 3.4-2 and 3.4-3.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of either RHR relief valve or an RCS vent opening of greater than or equal to 2.85 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 310°F. Either RHR relief valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures provided measures are taken to cushion the overpressure effects at RCS temperatures above 250°F, or (2) the start of 3 charging pumps and their injection into a water solid RCS.

3/4.4.11 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a(g)(6)(i).

3/4.4.12 REACTOR VESSEL HEAD VENTS

The OPERABILITY of the Reactor Head Vent System ensures that adequate core cooling can be maintained in the event of the accumulation of non-condensable gases in the reactor vessel. This system is in accordance with 10 CFR 50.44(c)(3)(iii).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 108 TO FACILITY OPERATING LICENSE NO. NPF-2

AND AMENDMENT NO. 100 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

On June 25, 1990, the staff issued Generic Letter (GL) 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,' pursuant to 10 CFR 50.54(f)." The Generic Letter represented the technical resolution of the above-mentioned generic issues.

Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability," involves the evaluation of the reliability of power-operated relief valves (PORVs) and block valves and their safety significance in PWR plants. The Generic Letter discussed how PORVs are increasingly being relied on to perform safety-related functions and the corresponding need to improve the reliability of both PORVs and their associated block valves. Proposed staff positions and improvements to the plant's Technical Specifications (TS) were recommended to be implemented at all affected facilities. This issue is applicable to all Westinghouse, Babcock & Wilcox, and Combustion Engineering designed facilities with PORVs.

Generic Issue 94, "Additional Low-Temperature Overpressure Protection for Light-Water Reactors," addresses concerns with the implementation of the requirements set forth in the resolution of Unresolved Safety Issue (USI) A-26, "Reactor Vessel Pressure Transient Protection (Overpressure Protection)." The Generic Letter discussed the continuing occurrence of overpressure events and the need to further restrict the allowed outage time for a low-temperature overpressure protection (LTOP) channel in operating modes 4, 5 and 6. This issue is only applicable to Westinghouse and Combustion Engineering facilities.

By letters dated May 13, 1991, and October 13, 1992, the Southern Nuclear Operating Company (SNC or the licensee) proposed changes to the TS in response to GL 90-06. Amendment No. 97 for Unit 1 and Amendment No. 89 for Unit 2, which addresses the requirements of GL 90-06 related to Generic Issue 70, were issued on March 8, 1993.

The Safety Evaluation in this amendment addresses the proposed TS changes submitted by SNC related to Generic Issue 94.

The actions proposed by the NRC staff to improve the availability of the LTOP system represents a substantial increase in the overall protection of the public health and safety and a determination has been made that the attendant costs are justified in view of this increased protection. The technical findings and the regulatory analysis related to Generic Issue 94 are discussed in NUREG-1326, "Regulatory Analysis for the Resolution of Generic Issue 94, Additional Low-Temperature Overpressure Protection for Light-Water Reactors."

In a letter dated May 13, 1991, SNC proposed changes to the LTOP TS to address the concerns of Generic Issue 94. In this letter, SNC agreed with the staff that the greatest risk of an overpressure event would occur during water solid operation. This conclusion was based on a Westinghouse plant-specific probabalistic risk assessment (PRA) performed with one residual heat removal (RHR) relief valve out of service (the RHR relief valves provide the LTOP protection for the RCS at Farley). The Westinghouse PRA ("Allowable Outage Time Study for Residual Heat Removal Valves for Farley Units 1 and 2," WCAP-12933) showed an approximate 54 percent reduction in core damage frequency can be realized by reducing the allowed outage time for an RHR relief valve from the current 7 days to 24 hours for water solid operation. Based on this assessment, SNC proposed a TS change that reduces the allowed outage time for an inoperable RHR relief valve with the RCS water solid from 7 days to 24 hours.

In Enclosure B to GL 90-06, the staff determined that the unavailability of LTOP protection is the dominant contributor to LTOP transients. The staff further concluded that during water solid operation, when the potential for an overpressure event is greatest, a substantial improvement in availability can be achieved through increased administrative restrictions. The staff has concluded that the SNC proposed TS significantly reduces the time Farley would be in a water solid condition when one RHR relief valve is out of service and it is consistent with the staff's conclusions contained in GL 90-06; therefore, the staff finds the proposed TS change is acceptable.

SNC also evaluated the risk from an overpressure event during operating Modes 5 and 6 when the RCS is not water solid and concluded, based on the Westinghouse analysis, that the reduction in risk realized from a more restrictive allowed outage time for an LTOP channel is not significant. As a result, SNC did not propose to modify the current 7-day LCO for an inoperable RHR relief valve for operation in non-water solid conditions. In a letter dated August 14, 1992, the staff provided the results of its review of the submittals related to GL 90-06. In this letter, the staff stated that SNC has modified the staff position with regard to Generic Issue 94 and that PRA based arguments to expand allowed outage times or modify generic letter requirements are not acceptable. However, the staff also stated that it would be receptive to extending the recommended 24-hour allowed outage time with an inoperable

LTOP channel to 7 days, provided the plant is not water solid and a level of protection comparable to that of a nitrogen bubble in Babcock and Wilcox plants is provided.

In a letter dated October 13, 1992, SNC proposed to revise the limiting condition for operation for TS Section 3.4.10.3 and its corresponding Bases Section for the Joseph M. Farley Nuclear Plant, Units 1 and 2, as follows:

1. Revise Limiting Condition for Operation (LCO) Action Statement "a" for Technical Specification 3.4.10.3 to reduce the allowed outage time for one RHR relief valve from the current 7 days to 24 hours unless: 1) the pressurizer water level is reduced to equal to or less than 30 percent (cold calibrated), and 2) a dedicated operator is assigned to perform RCS pressure monitor and control functions.

2. Revise the Bases Section for Technical Specification 3/4.4.10 to clarify the means of providing low-temperature overpressure protection for the limiting heat addition transient.

Section 5.2.2 of the Updated Final Safety Analysis Report states that the reactor coolant system (RCS) LTOP is provided during startup and shutdown when the RCS is in a water solid condition by two independent RHR suction relief valves. In its October 13, 1992, submittal, SNC stated that the Joseph M. Farley LTOP system and the supporting analysis is based on the fact that there is sufficient capacity provided by one RHR relief valve to limit the effects of: (1) the worst case mass input transient (inadvertent start of charging pumps), and (2) the limiting heat addition transient (reactor coolant pump (RCP) start) provided measures are taken to cushion the overpressure effects at RCS temperatures above 250°F.

In response to the staff's comments in its August 14, 1992, letter, SNC selected a pressurizer level of 30 percent (cold calibrated) as the definition of water solid conditions. This level was chosen to allow the operator sufficient time to respond to the overpressure event so that the limits of Appendix G are not violated. In addition, to evaluate the risk while operating under an LCO for one inoperable LTOP channel, the postulated failure of the other LTOP channel was considered.

An analysis of the consequences of the inadvertent start of two charging pumps (assuming both LTOP channels are inoperable with no other RCS vents available) and an initial pressurizer level of 30 percent was performed that predicted the limits of Appendix G would be exceeded within approximately 3.5 minutes. It should be noted that the results of the prior analysis contained in the SNC May 13, 1991, submittal of the limiting heat addition transient resulting from the start of an RCP with a temperature difference between the steam generators and the RCS primary side of less than 50°F concluded that an initial pressurizer level of 30 percent provides sufficient capacity for water expansion to prevent the limits of Appendix G from being exceeded.

To provide assurance that the overpressure protection system is not challenged during the 7-day allowed outage time due to the short period of time in which an operator must respond to an inadvertent charging pump start, SNC proposed a dedicated operator to monitor and control the RCS pressure whenever an RHR suction relief valve is inoperable. SNC has also stated that Farley has two independent alarms to protect against a low temperature over-pressurization event, a low temperature over-pressurization alarm set at 425 pounds per square inch (psi) (the RHR relief valve setpoint is 450 psi) and a high pressurizer level alarm set at the 75 percent pressurizer level.
SNC did not follow the proposed guidance contained in Enclosure B to GL 90-06 because the switches in the control room that operate the charging pumps do not have a pull-to-lock feature. As a result, the automatic initiation mode for the charging pumps cannot be bypassed from the control room. Isolation can be achieved by securing the pump motor circuit breaker in the open position at the motor control center (MCC). Since these pumps also cool the reactor coolant pump seals, the licensee has stated it is reluctant to put the plant in a condition where the failure of the operating charging pump could result in damage to the RCP seals because of the increased time it would take to start one of the charging pumps if power was removed at the MCC.

The staff has reviewed SNC proposed modifications to the TS, and because SNC has proposed a trained dedicated operator to monitor and control RCS pressure, the staff has determined that reasonable assurance exists that this operator can take timely corrective actions to mitigate an LTOP event. This conclusion is based on the fact that two alarms are available to identify the occurrence of an overpressure event and the operator has been specifically trained to respond to these alarms. Although it is not likely that the operator will detect and respond to an LTOP event prior to receiving an alarm because the event can be terminated by the action of tripping the charging pumps or closing a valve, the trained operator should be able to perform one of these simple actions prior to the overpressurization of the reactor vessel occurring. Based on the above evaluation, the staff finds the proposed changes to the TS to mitigate an LTOP event when the reactor primary system is not in a water solid condition to be acceptable.

On the basis of the review of the SNC submittals, the staff has determined that the TS changes proposed by the licensee meet the intent of the requirements of GL 90-06 with regard to Generic Issue 94 for the Farley Nuclear Plants. With the resolution of Generic Issue 94, GL 90-06 is considered closed.

3.0 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.

4.0 ENVIRONMENTAL CONCLUSION

The amendment involves a change in a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. The 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 exposures. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and that there has been no public comment on such findings (58 FR 7005 and 58 FR 8787). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 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.

5.0 CONCLUSION

The staff 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, and (2) such activities will be conducted in compliance with the commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public. The staff, therefore, concludes that the proposed changes are acceptable.

Principal Contributors: Edward Throm
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