September 3, 1992

Docket No. 50-443

Mr. Ted C. Feigenbaum, Senior Vice President
and Chief Nuclear Officer
North Atlantic Energy Service Corporation
Post Office Box 300
Seabrook, New Hampshire 03874

Dear Mr. Feigenbaum:

SUBJECT: ISSUANCE OF AMENDMENT NO. 15 TO FACILITY OPERATING LICENSE NO. NPF-NPF-86 - SEABROOK STATION, UNIT NO. 1 (TAC NO. M83346)

The Commission has issued the enclosed Amendment No. 15 to Facility Operating License No. NPF-86 for the Seabrook Station. This amendment is in response to your application dated May 5, 1992.

This amendment changes the Seabrook Station Technical Specifications to allow a relaxation in the pressurizer safety valve (PSV) and main steam safety valve (MSSV) setpoint tolerances to ±3%. The ±3% tolerance is used for the "as found" acceptance criteria for additional valve testing.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly FEDERAL REGISTER notice.

Sincerely,

Original signed by
Gordon E. Edison, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 15 to License No. NPF-86
2. Safety Evaluation

cc w/enclosures:
See next page

8-21-92
EMEB: DET

LA:PDI-3	INT:PDI-3	PM:PDI-3	OGC	D:PDI-3	OTSB
TClark	KBattige:sk	GEdison		WButler	CGrimes
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based on MEB review.

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

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and Chief Nuclear Officer
North Atlantic Energy Service Corporation
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G E Edison

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

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   License No. NPF-86
2. Safety Evaluation

cc w/enclosures:
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AMENDMENT NO. 15 TO NPF-86 SEABROOK STATION DATED September 3, 1992

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NORTH ATLANTIC ENERGY SERVICE CORPORATION, ET AL.*

DOCKET NO. 50-443

SEABROOK STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No15
License No. NPF-86

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

   A. The application for amendment filed by the North Atlantic Energy Service Corporation (NAESCO) (the licensee), acting for itself and as agent and representative of the 11 other utilities listed below and hereafter referred to as licensees, dated May 5, 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 set forth in 10 CFR Chapter I;

   D. The issuance of this 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.

*North Atlantic Energy Service Corporation is authorized to act as agent for the North Atlantic Energy Corporation, the Canal Electric Company, The Connecticut Light and Power Company, EUA Power Corporation, Hudson Light & Power Department, Massachusetts Municipal Wholesale Electric Company, Montaup Electric Company, New England Power Company, New Hampshire Electric Cooperative, Inc., Taunton Municipal Light Plant, The United Illuminating Company, and Vermont Electric Generation and Transmission Cooperative, Inc., and has exclusive responsibility and control over the physical construction, operation and maintenance of the facility.

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-86 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 15, and the Environmental Protection Plan contained in Appendix B are incorporated into Facility License No. NPF-86. NAESCO shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of receipt of this letter.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter Butler, Director
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 3, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 15

FACILITY OPERATING LICENSE NO. NPF-86

DOCKET NO. 50-443

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. *Overlap pages have been provided.

Remove	Insert
3/4 4-7*	3/4 4-7*
3/4 4-8	3/4 4-8
3/4 4-9	3/4 4-9
3/4 4-10*	3/4 4-10*
3/4 7-1*	3/4 7-1*
3/4 7-2	3/4 7-2
B 3/4 7-1	B 3/4 7-1
B 3/4 7-2*	B 3/4 7-2*

REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.2 Two residual heat removal (RHR) loops shall be OPERABLE* and at least one RHR loop shall be in operation.**

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

a. With less than the above required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.2 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

*One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

**The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer Code safety valve shall be OPERABLE with a lift setting* of 2485 psig ± 3%.**

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer Code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

SURVEILLANCE REQUIREMENTS

4.4.2.1 No additional requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.
**Within ± 1% following pressurizer Code safety valve testing

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting* of 2485 psig ± 3%.**

APPLICABILITY: MODES 1, 2, and 3#.

ACTION:

With one pressurizer Code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

**Within ± 1% following pressurizer Code safety valve testing

#Entry into this MODE is permitted for up to 24 hours to perform post-modification or post-maintenance testing to verify OPERABILITY of components. ACTION requirements shall not apply until OPERABILITY has been verified.

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a water volume of less than or equal to 92% of pressurizer level (1656 cubic feet), and at least two groups of pressurizer heaters each having a capacity of at least 150 kW.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With only one group of pressurizer heaters OPERABLE, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the Reactor Trip System breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The pressurizer water volume shall be determined to be within its limit at least once per 12 hours.

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by energizing the heaters from the emergency power supply and measuring circuit current at least once per 92 days.

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line Code safety valves associated with each steam generator shall be OPERABLE with lift settings as specified in Table 3.7-2.

APPLICABILITY: MODES 1, 2, and 3#.

ACTION:

With four reactor coolant loops and associated steam generators in operation and with one or more main steam line Code safety valves inoperable, operation in MODES 1, 2, and 3 may proceed, provided that within 4 hours either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional requirements other than those required by Specification 4.0.5.

#Entry into this MODE is permitted for up to 24 hours to perform post-modification or post-maintenance testing to verify OPERABILITY of components. ACTION requirements shall not apply until OPERABILITY has been verified.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING FOUR-LOOP OPERATION

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)
1	87
2	65
3	43

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER					
Loop 1	Loop 2	Loop 3	Loop 4	LIFT SETTING* (± 3%)**	ORIFICE SIZE
V6	V22	V36	V50	1185 psig	16.0 sq. in.
V7	V23	V37	V51	1203 psig	16.0 sq. in.
V8	V24	V38	V52	1220 psig	16.0 sq. in.
V9	V25	V39	V53	1238 psig	16.0 sq. in.
V10	V26	V40	V54	1255 psig	16.0 sq. in.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

**Within ± 1% following main steam line Code safety valve testing

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line Code safety valves ensures that the Secondary System pressure will be limited to within 110% (1320 psia) of its design pressure of 1200 psia during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a Turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, (1974 Edition, including the Summer 1975 Addenda). The total relieving capacity for all valves on all of the steam lines is 1.839×10^7 lbs/hr which is 121% of the total secondary steam flow of 1.514×10^7 lbs/hr at 100% RATED THERMAL POWER. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in Secondary Coolant System steam flow and THERMAL POWER required by the reduced Reactor trip settings of the Power Range Neutron Flux channels. The Reactor Trip Setpoint reductions are derived on the following bases:

For four loop operation:

$$SP = \frac{(X) - (Y)(V)}{X} \times 109$$

Where:

SP = Reduced Reactor Trip Setpoint in percent of RATED THERMAL POWER,

V = Maximum number of inoperable safety valves per steam line,

109 = Power Range Neutron Flux-High Trip Setpoint for four loop operation,

X = Total relieving capacity of all safety valves per steam line in lbs/hr, and

Y = Maximum relieving capacity of any one safety valve in lbs/hr

3/4.7.1 TURBINE CYCLE (Continued)

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the Auxiliary Feedwater System ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss-of-offsite power.

The electric motor-driven emergency feedwater pump is capable of delivering a total feedwater flow of 650 gpm at a pressure of 1221 psig to the entrance of the steam generators. The steam-driven emergency feedwater pump is capable of delivering a total feedwater flow of 650 gpm at a pressure of 1221 psig to the entrance of the steam generators. The startup feedwater pump serves as the third auxiliary feedwater pump and can be manually aligned to be powered from an emergency bus (Bus 5). The startup feedwater pump is capable of taking suction on the dedicated emergency feedwater volume of water in the condensate storage tank and delivering a total feedwater flow of in excess of 650 gpm at a pressure of 1221 psig to the entrance of the steam generator via either the main feedwater header or with manual alignment to the emergency feedwater flow path. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the Residual Heat Removal System may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to cool the RCS to a temperature of 350°F. The OPERABILITY of the concrete enclosure ensures this availability of water following rupture of the condensate storage tank by a tornado generated missile. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 SPECIFIC ACTIVITY

The limitations on Secondary Coolant System specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of a steam line rupture. This dose also includes the effects of a coincident 1 gpm reactor-to-secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to: (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the Surveillance Requirements are consistent with the assumptions used in the safety analyses.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 15 TO FACILITY OPERATING LICENSE NO. NPF-86

NORTH ATLANTIC ENERGY SERVICE CORPORATION

SEABROOK STATION, UNIT NO. 1

DOCKET NO. 50-443

1.0 INTRODUCTION

By letter dated May 5, 1992, the Public Service Company of New Hampshire (former licensee) submitted a request for changes to the Seabrook Station Technical Specifications (TSs). Public Service Company of New Hampshire has transferred management authority for Seabrook Station to North Atlantic Energy Service Corporation (current licensee). The proposed change would allow relaxation of the Pressurizer Safety Valve (PSV) and the Main Steam Safety Valve (MSSV) setpoint tolerance from ±1% to ±3%. The licensee proposes to use the ±3% tolerance for the "as-found" acceptance criteria for additional valve testing required by ASME Section XI, Article IWV-3513. The proposed TS changes require that the PSV and MSSV setpoints be restored to within ±1% of their nominal setpoints following testing. The licensee also proposes to revise the Bases for TS 3/4.7.1.1 to specify the correct Edition of the ASME Boiler and Pressure Vessel Code, Section III applicable to the MSSVs. The proposed relaxation of the PSV and MSSV setpoint tolerances complies with the 1989 Edition of the ASME Code, Section XI and has been evaluated by Yankee Atomic Electric Company (YAEC) (YAEC-1847, "Seabrook Station Code Safety Valve Setpoint Tolerance Relaxation," February 28, 1992).

2.0 EVALUATION

The Seabrook Station overpressure protection design incorporates three Code safety valves on the primary system pressurizer and a total of 20 Code safety valves on the four main steam lines (five per line) in the secondary system. The pressurizer safety valves (PSVs) were designed and manufactured to meet the 1971 Edition, including the Winter 1972 Addenda, of the ASME Code, Section III. The main steam safety valves (MSSVs) were designed and manufactured to meet the 1974 Edition, including the Summer 1975 Addenda, of the ASME Code, Section III. An ASME Code, Section III requirement for both the PSVs and MSSVs is that they be designed to open within ±1% of the set pressure. The current TS Limiting Condition for Operation (LCO) for the PSVs and MSSVs also imposes the tolerance of ±1% on their set pressure.

Currently the TS Surveillance Requirements for the PSVs and the MSSVs requires that testing be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda, as required by 10 CFR 50.55a(g). The surveillance requirements indicate that the MSSVs and PSVs should be tested to verify that their lift pressure and seat leakages are acceptable pursuant to the Seabrook Inservice Test (IST) program which complies with the

ASME Boiler and Pressure Vessel Code, Section XI, 1983 Edition, through the Summer 1983 Addenda. This document does not indicate the tolerance to be applied to the safety valve lift pressure verification. The licensee therefore uses the ±1% indicated in the LCO as the acceptance criteria for the PSVs and MSSVs during ASME Section XI testing. Under the current testing requirements, when a PSV or MSSV has a tested lift pressure outside the ±1% tolerance specified in the LCO, it must be repaired or replaced and additional valves in the system must be tested.

The 1989 Edition of the ASME Code, Section XI, now requires that the PSVs and MSSVs be tested pursuant to the ASME/ANSI OM-1987, Part 1, "Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices." This allows the tested lift pressure to exceed the stamped set pressure by up to 3% before declaring a test failure. It also includes guidelines for testing additional valves when a valve exceeds the ±3% tolerance. Therefore, increasing the PSV and MSSV setpoint tolerance to ±3% for testing acceptance criteria is in compliance with the later ASME Code, Section XI requirements.

The licensee proposes to use the ±3% tolerance for the "as-found" acceptance criteria for additional valve testing required by ASME Section XI Subsection IWV-3513. The proposed TS revisions require that PSV and MSSV Setpoints be restored to within ±1% of their nominal setpoints following testing.

The 1989 Edition of the ASME Code Section III, Subarticle NB-7410/NC-7410 states that "The set pressure of at least one of the pressure relief devices connected to the system not be greater than the Design Pressure of any component within the pressure retaining boundary of the protected system". The Reactor Coolant System design pressure is 2485 psig, which corresponds to the setpoint of the PSVs. The Main Steam Supply System design pressure is 1185 psig, which corresponds to the Group 1 MSSVs which have the lowest opening setpoint. Therefore, the proposed relaxation of the setpoint tolerances for the PSVs and the MSSVs has been determined to be in compliance with this edition of the Code.

The licensee evaluated each of the transients considered in the Updated Final Safety Analysis Report (UFSAR) to determine the effects of increased safety valve setpoint tolerance.

The UFSAR Departure from Nucleate Boiling Ratio (DNBR) evaluations take credit for operation of the pressurizer PORVs, which have a setpoint pressure of 2400 psia. Since this is lower than the proposed lower limit of 2425 psia on PSV setpoint, revising the safety valve setpoint tolerance does not affect DNBR. Increasing pressure yields less limiting values of DNBR. Therefore, all further evaluations address peak pressure response.

A margin of 24 psi remains between the Atmospheric Steam Dump Valve (ASDV) setpoints and the relaxed Group 1 MSSV setpoints. The Group 1 MSSVs, which have the lowest opening setpoint, have a proposed lower limit setpoint (-3%

tolerance) of 1164 psia. This is sufficiently above the ASDV opening setpoint of 1140 psia to prevent unnecessary challenges to the MSSVs. The licensee also concluded that the proposed lower limit of the PSV setpoint, 2425 psia, will not affect the automatic reactor trip on high pressurizer pressure which occurs at 2400 psia.

2.1 Turbine Trip

A detailed analysis was performed for the most limiting pressurization transient, the turbine trip. This event was simulated using the RETRAN01 MOD5 computer code. Comparisons were performed with the licensing basis analysis of record to demonstrate that the RETRAN model provides comparable results. The limiting peak pressure case with minimum reactivity feedback and without pressure control was simulated using RETRAN. The RETRAN results slightly overpredicted the peak pressure compared to the UFSAR. Evaluations with revised setpoints were then performed with the RETRAN model to demonstrate that the peak pressure remains well below the Condition II limit of 110% of design pressure, or 2750 psia for the primary system and 1320 psia for the secondary system.

2.2 LOCA Analysis

The postulated large break loss of coolant accident (LBLOCA) events in the Seabrook UFSAR do not challenge the MSSVs because the primary system pressure draws down the secondary pressure almost immediately after initiation. The postulated small break LOCA (SBLOCA) events, however, by virtue of their break sizes (less than 1 ft^2), can challenge the MSSVs because the secondary side plays the role of the heat sink early in the transient. Since the PSVs are not challenged in a LOCA transient, the proposed PSV setpoint tolerance relaxation does not affect LOCA analyses. Therefore, the LOCA review will only include the effects of SBLOCA on the MSSVs.

The UFSAR SBLOCA defines the limiting event as a cold leg pipe rupture of an equivalent 4-inch diameter at the ECCS injection location. The UFSAR evaluation of the limiting SBLOCA event yields a Peak Cladding Temperature (PCT) of 1790.0°F. The current SBLOCA PCT including margin allocation is 1973°F. The licensee's evaluation of the Code safety valve setpoint tolerance relaxation specifies an increase in the limiting SBLOCA PCT of about 2.5°F. The licensee recommends a conservative PCT penalty of 5°F be applied to the SBLOCA PCT results and be tracked in accordance with 10 CFR 50.46 reporting requirements. Since this increase in PCT is less than 50°F, it is not considered a significant change as defined by 10 CFR 50.46. The revised SBLOCA PCT value of 1978.2°F remains below the 2200°F limit. Pending approval of this license amendment, the licensee has committed to report the SBLOCA PCT increase in its annual 10 CFR 50.46 report.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations and past practice, the New Hampshire and Massachusetts State officials were notified of the proposed issuance of the amendment. The State officials had no comments.

4.0 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 and changes surveillance requirements. 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 (57 FR 24677). 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.

5.0 CONCLUSION

The staff has reviewed the licensee's evaluation on the impacts of the proposed TS changes, TS 3/4.4.2 and TS Table 3.7-2, to allow a relaxation in the PSV and MSSV setpoint tolerances to ± 3% for testing acceptance criteria. The evaluation concluded that (1) there will be no reduction in the calculated minimum DNBR, (2) overpressurization events do not exceed the safety limits, and (3) the acceptance criteria for ECCS performance were not exceeded for LOCA events. The staff finds the evaluation and conclusions acceptable.

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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