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

September 21, 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.16TO FACILITY OPERATING LICENSE NO. NPF-86, SEABROOK STATION, UNIT NO. 1 AND CLOSEOUT OF NRC GENERIC LETTER NO. 90-06 (TAC NOS. M77377 AND M77452)

The Commission has issued the enclosed Amendment No. 16 to Facility Operating License No. NPF-86 for the Seabrook Station, Unit 1. The amendment is in response to your application dated October 16, 1991 which supplemented your response to generic letter 90-06 dated December 21, 1990.

Generic Letter 90-06, dated June 25, 1990, required pressurized water reactor licensees to revise their Technical Specifications (TS), to improve the availability of power operated relief valves (PORVs), block valves and Overpressure Protection Systems which may be used to mitigate Reactor Coolant System (RCS) pressure transients. These TS changes implement the requirements of Generic Issue 70 (Multi-Plant Action (MPA) B-114), "Power Operated Relief Valve and Block Valve Reliability", and Generic Issue 94 (MPA B-115), "Additional Low Temperature Overpressure Protection for Light Water Reactors."

This amendment revises the TS for Seabrook Station, Unit No. 1, implementing the guidance of Generic Letter 90-06 involving enhanced safe plant operation by improving the availability of PORVs and block valves and Overpressure Protection Systems which may be used to mitigate RCS pressure transients.

You responded to NRC Generic Letter 90-06 for the Seabrook Station, by your letter number NYN-90217, dated December 21, 1990. In NYN-90217, you indicated that you intended to submit changes to Seabrook Station TS implementing the guidance of Generic Letter 90-06. You submitted proposed changes to the Seabrook Station TS by your letter number NYN-91167, dated October 16, 1991. In NYN-91167, you indicated that your proposed TS changes are based upon model TS provided with Generic Letter 90-06. You reported on the status of Generic Letter 90-06 during the first Seabrook Station refueling outage in your letter number NYN-91180, dated November 4, 1991. In NYN-91180, you indicated that your action regarding Generic Letter 90-06 was complete and that no further action by you was required.

Based on our review of your responses to Generic Letter 90-06, we conclude that the actions taken to date fully respond to the issues identified in the subject generic letter. Therefore, we consider this issue and TAC Nos. M77377 and M77452 closed by this letter. The staff may conduct an inspection of your implementation at a later time.

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

Sincerely,

/S/

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

Enclosures:
1. Amendment No. 16 to
   License No. NPF-86
2. Safety Evaluation

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DATE	9/18/92		/ /	/ /	/ /

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

Sincerely,

G E Edison

Gordon E. Edison, Sr. Project Manager
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Enclosures:
1. Amendment No. 16 to
   License No. NPF-86
2. Safety Evaluation

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AMENDMENT NO. 16 TO NPF-86 SEABROOK STATION DATED September 21, 1992

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

NORTH ATLANTIC ENERGY SERVICE CORPORATION, ET AL.*

DOCKET NO. 50-443

SEABROOK STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 16
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 (the licensee), acting for itself and as agent and representative of the 11 other utilities listed below and hereafter referred to as licensees, dated October 16, 1991, 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. 16 , 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 R. 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 21, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 16

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 are provided for continuity.

Remove	Insert
3/4 4-11	3/4 4-11
3/4 4-12	3/4 4-12
3/4 4-34	3/4 4-34
3/4 4-34a	3/4 4-34a
3/4 4-35	3/4 4-35
B 3/4 4-2	B 3/4 4-2
---	B 3/4 4-2a*
B 3/4 4-15	B 3/4 4-15

*Denotes new page

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 Both power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or both PORV(s) inoperable, because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one PORV inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV to OPERABLE status or close the associated block valve and remove power from the block valve; restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With both PORVs inoperable due to causes other than excessive seat leakage, within 1 hour either restore at least one PORV to OPERABLE status or close each associated block valve and remove power from the block valve and be in HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

d. With one or both block valves inoperable, within 1 hour restore the block valve(s) to OPERABLE status or place its associated PORV(s) control switch to "CLOSE". Restore at least one block valve to OPERABLE status within the next hour if both block valves are inoperable; restore any remaining inoperable block valve to operable status within 72 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

e. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by:

a. Performance of a CHANNEL CALIBRATION, and

b. Operating the valve through one complete cycle of full travel during MODES 3 or 4.

4.4.4.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed with power removed in order to meet the requirements of ACTION b. or c. in Specification 3.4.4.

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 200°F in any 1-hour period, and

c. A maximum spray water temperature differential of 320°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

REACTOR COOLANT SYSTEM

PRESSURE/TEMPERATURE LIMITS

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 The following Overpressure Protection Systems shall be OPERABLE:

a. In MODE 4 when the temperature of any RCS cold leg is less than or equal to 329°F; and in MODE 5 and MODE 6 with all Safety Injection pumps inoperable at least one of the following groups of two overpressure protection devices shall be OPERABLE when the RCS is not depressurized with an RCS vent area of greater than or equal to 1.58 square inches:

   1) Two residual heat removal (RHR) suction relief valves each with a setpoint of 450 psig +0, -3 %; or

   2) Two power-operated relief valves (PORVs) with lift setpoints that vary with RCS temperature which do not exceed the limit established in Figure 3.4-4, or

   3) One RHR suction relief valve and one PORV with setpoints as required above.

b. In MODE 5 and MODE 6 with all Safety Injection pumps except one inoperable:

   1) The Reactor Coolant System (RCS) depressurized with an RCS vent area equal to or greater than 18 square inches.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 329°F; MODE 5 and MODE 6 with the reactor vessel head on.

ACTION:

a) In MODE 4 with all Safety Injection pumps inoperable and with one of the two required overpressure protection devices inoperable, either restore two overpressure protection devices to OPERABLE status within 7 days or within the next 8 hours

   (a) depressurize the RCS and

   (b) vent the RCS through at least a 1.58-square-inch vent.

b) In MODE 5 and MODE 6 with all Safety Injection pumps inoperable and with one of the two required overpressure protection devices inoperable, restore two overpressure protection devices to OPERABLE status within 24 hours or within the next 8 hours

   (a) depressurize the RCS and

   (b) vent the RCS through at least a 1.58-square-inch vent.

ACTION: (Continued)

c) In MODE 4, MODE 5 and MODE 6 with all Safety Injection pumps inoperable and with both of the two required overpressure protection devices inoperable, within the next 8 hours

(a) depressurize the RCS and

(b) vent the RCS through at least a 1.58-square-inch vent.

d) In the event the PORVs, or the RHR suction relief valves, or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.8.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs, or the RHR suction relief valves, or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

e) In MODE 5 and MODE 6 with all Safety Injection pumps except one inoperable and with the RCS vent area less than 18 square inches, immediately restore all Safety Injection pumps to inoperable status.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE when the PORV(s) are being used for overpressure protection by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, at least once per 31 days thereafter when the PORV is required OPERABLE; and

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and

c. Verifying the PORV isolation valve is open at least once per 72 hours.

4.4.9.3.2 Each RHR suction relief valve shall be demonstrated OPERABLE when the RHR suction relief valve(s) are being used for overpressure protection as follows:

a. For RHR suction relief valve RC-V89 by verifying at least once per 72 hours that RHR suction isolation valves RC-V87 and RC-V88 are open.

b. For RHR suction relief valve RC-V24 by verifying at least once per 72 hours that RHR suction isolation valves RC-V22 and RC-V23 are open.

c. Testing pursuant to Specification 4.0.5.

4.4.9.3.3 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

*Except when the vent pathway is provided with a valve(s) or devices(s) that is locked, sealed, or otherwise secured in the open position, then verify this valve(s) or device(s) open at least once per 31 days.

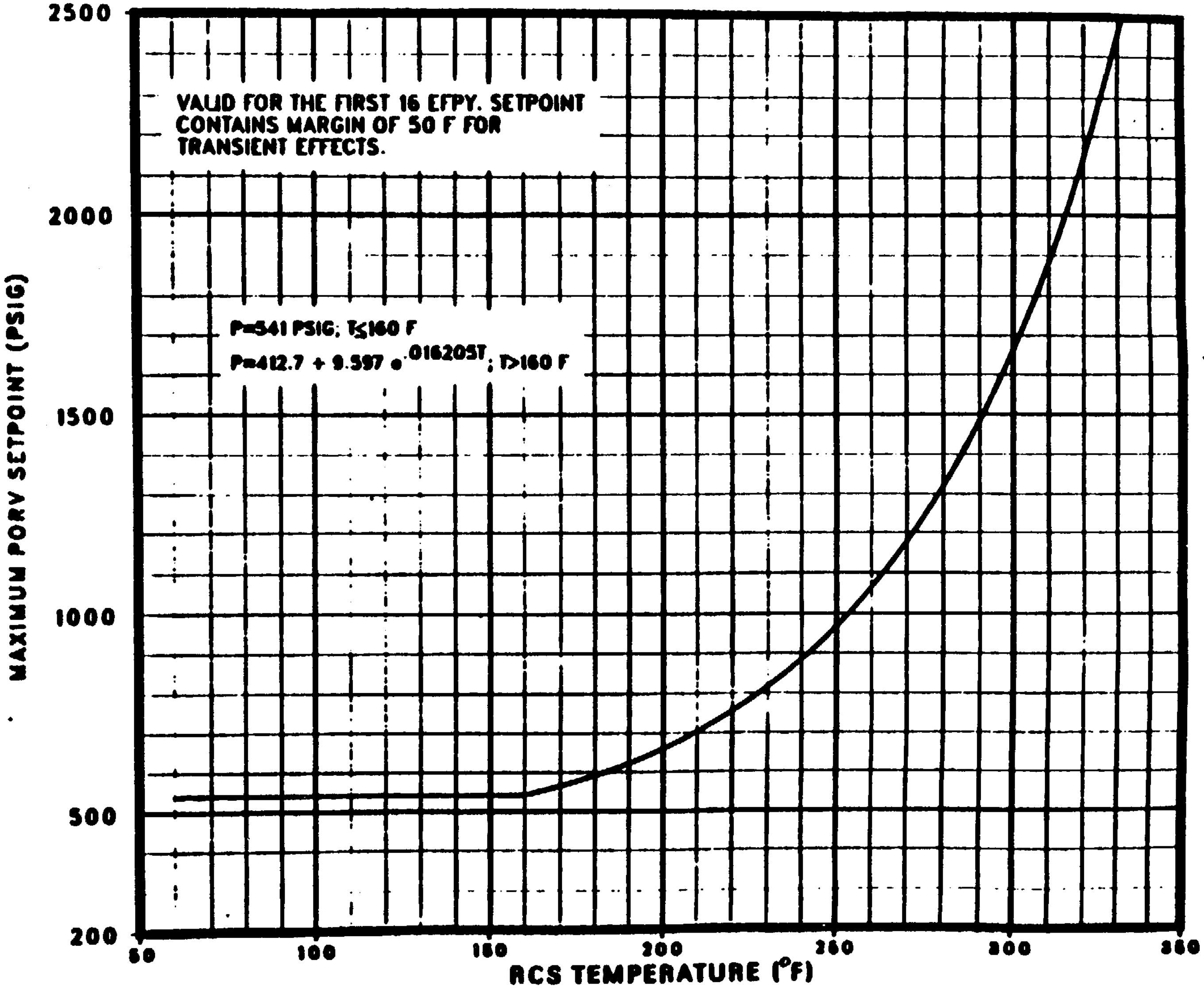

FIGURE 3.4-4 RCS COLD OVERPRESSURE PROTECTION SETPOINTS

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above 1.30 during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 6 hours.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing core decay heat even in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal capacity if a bank withdrawal accident can be prevented, i.e., by opening the Reactor Trip System breakers. Single failure considerations require that two loops be OPERABLE at all times.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR loops be OPERABLE.

The operation of one reactor coolant pump (RCP) or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting an RCP in MODES 4 and 5 are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold-leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve Setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

3/4.4.2 SAFETY VALVES (Continued)

During operation, all pressurizer Code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no Reactor trip until the first Reactor Trip System Trip Setpoint is reached (i.e., no credit is taken for a direct Reactor trip on the loss of load) and also assuming no operation of the power-operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.3 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady-state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12-hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

3/4.4.4 RELIEF VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer Code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable. The PORVs and their associated block valves are powered from Class 1E power supply busses.

The PORVs are equipped with automatic actuation circuitry and manual control capability. The PORVs are considered OPERABLE in either the automatic or manual mode for the following reasons:

(1) No credit is taken in any FSAR accident analysis for automatic PORV actuation to mitigate the consequences of an accident.

(2) No Surveillance Requirement (ACOT or TADOT) exists for verifying automatic operation.

(3) The required ACTION for an inoperable PORV(s) (closing the block valve) conflicts with any presumed requirement for automatic actuation.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation, so that corrective measures can be taken.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

COLD OVERPRESSURE PROTECTION

The OPERABILITY of two PORVs, or two RHR suction relief valves, or a combination of a PORV and RHR suction relief valve, or an RCS vent opening of at least 1.58 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 329°F. Either PORV or either RHR suction 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, or (2) the start of a centrifugal charging pump and its injection into a water-solid RCS.

The Maximum Allowed PORV Setpoint for the Cold Overpressure Mitigation System (COMS) is derived by analysis which models the performance of the COMS assuming various mass input and heat input transients. Operation with a PORV Setpoint less than or equal to the maximum Setpoint ensures that Appendix G criteria will not be violated with consideration for: (1) a maximum pressure overshoot beyond the PORV Setpoint which can occur as a result of time delays in signal processing and valve opening; (2) a 50°F heat transport effect made possible by the geometrical relationship of the RHR suction line and the RCS wide range temperature indicator used for COMS; (3) instrument uncertainties; and (4) single failure. To ensure mass and heat input transients more severe than those assumed cannot occur, Technical Specifications require lockout of both Safety Injection pumps and all but one centrifugal charging pump while in MODES 4, 5, and 6 with the reactor vessel head installed and disallow start of an RCP if secondary coolant temperature is more than 50°F above reactor coolant temperature. Exceptions to these requirements are acceptable as described below.

Operation above 350°F but less than 375°F with only centrifugal charging pump OPERABLE and no Safety Injection pumps OPERABLE is allowed for up to 4 hours. As shown by analysis, LOCAs occurring at low temperature, low pressure conditions can be successfully mitigated by the operation of a single centrifugal charging pump and a single RHR pump with no credit for accumulator injection. Given the short time duration and the condition of having only one centrifugal charging pump OPERABLE and the probability of a LOCA occurring during this time, the failure of the single centrifugal charging pump is not assumed.

Operation below 350°F but greater than 325°F with all centrifugal charging and Safety Injection pumps OPERABLE is allowed for up to 4 hours. During low pressure, low temperature operation all automatic Safety Injection actuation signals except Containment Pressure - High are blocked. In normal conditions, a single failure of the ESF actuation circuitry will result in the starting of at most one train of Safety Injection (one centrifugal charging pump, and one Safety Injection pump). For temperatures above 325°F, an overpressure event occurring as a result of starting two pumps can be successfully mitigated by

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

COLD OVERPRESSURE PROTECTION (Continued)

operation of both PORVs without exceeding Appendix G limit. A single failure of a PORV is not assumed due to the short duration that this condition is allowed and the low probability of an event occurring during this interval in conjunction with the failure of a PORV to open. Initiation of both trains of Safety Injection during this 4-hour time frame due to operator error or a single failure occurring during testing of a redundant channel are not considered to be credible accidents.

Operation with all centrifugal charging pumps and both Safety Injection pumps OPERABLE is acceptable when RCS temperature is greater than 350°F, a single PORV has sufficient capacity to relieve the combined flow rate of all pumps. Above 350°F two RCPs and all pressure safety valves are required to be OPERABLE. Operation of an RCP eliminates the possibility of a 50°F difference existing between indicated and actual RCS temperature as a result of heat transport effects. Considering instrument uncertainties only, an indicated RCS temperature of 350°F is sufficiently high to allow full RCS pressurization in accordance with Appendix G limitations. Should an overpressure event occur in these conditions, the pressurizer safety valves provide acceptable and redundant overpressure protection.

When operating below 200°F in MODE 5 or MODE 6, Technical Specification 3.5.3.2 allows one Safety Injection pump to be made OPERABLE whenever the RCS has a vent area equal to or greater than 18 square inches. Cold overpressure protection in this configuration is provided by the 18 square inch or greater mechanical opening in the RCS pressure boundary. This mechanical opening is larger in size than the 1.58 square inch opening required for normal overpressure protection and is of sufficient size to ensure that the Appendix G limits are not exceeded when an SI pump is operating in MODE 5 or MODE 6. Additionally, when operating in a reduced inventory condition, the larger vent area limits RCS pressure during overpressure transients to reduce the possibility of adversely affecting steam generator nozzle dams. When the reactor has been shut down for at least 7 days, the larger vent area also enhances the ability to provide a gravity feed to the RCS from the Refueling Water Storage Tank in the unlikely event that the CCP and SI pumps were unavailable after a loss of RHR.

The Maximum Allowed PORV Setpoint for the Cold Overpressure Mitigation System will be revised on the basis of the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50, Appendix H.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 16 TO FACILITY OPERATING LICENSE NO. NPF-86

NORTH ATLANTIC ENERGY SERVICE CORPORATION

SEABROOK STATION, UNIT NO. 1

DOCKET NO. 50-443

1.0 INTRODUCTION

On June 25, 1990, the staff issued Generic Letter 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 pressurizer power operated relief valves (PORVs) and block valves and their safety significance in pressurized water reactor 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 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 channel in operating MODES 4, 5, and 6. This issue is only applicable to Westinghouse and Combustion Engineering facilities.

2.0 LICENSEE RESPONSES

New Hampshire Yankee (NHY), the former Seabrook licensee, responded to NRC Generic Letter 90-06 by letter number NYN-90217, dated December 21, 1990, for Seabrook Station, Unit 1. In NYN-90217, NHY indicated that two PORVs and associated block valves are installed in Seabrook Station. The Seabrook Station PORVs are solenoid-valve-controlled, pressure actuated, poppet-type relief valves which do not depend upon control air systems for their operation. The block valves are flexible wedge, rising stem, motor-operated gate valves.

The Seabrook Station PORVs, block valves and associated components were designed to meet safety-grade requirements. Mechanically, the PORVs and block valves are Safety Class 1 components. Electrically, the PORV solenoids and the block valve motor-operators are designated Class 1E. The PORV and block valve control switches, selector switches and position indications are also designated Class 1E. Other electrical components in the PORV and block valve circuitry including cables, connectors and splices that are located in harsh environments are environmentally qualified. The PORV circuitry includes contracts from relays driven by non-safety-related signals associated with automatic control functions. These relays are not qualified but are similar to Class 1E relays. An analysis has been performed to demonstrate that no credible failure of these relays can create a condition which degrades the function of the Class 1E portion of the PORV circuitry.

The PORVs and block valves at Seabrook Station are included within the scope of the NHY (now NAESCO) Operational Quality Assurance Program and are, therefore, listed in Updated Final Safety Analysis Report (FSAR) Table 3.2-2. Approved procedures implement maintenance requirements for the PORVs and block valves in accordance with manufacturer's recommendations. Maintenance is performed by trained maintenance personnel. Complete replacement PORVs or block valves would be purchased, if needed, in accordance with the requirements of the original construction specification including applicable amendments. Replacement parts for the PORVs or block valves would be purchased, if needed, by reference to the original manufacturer or supplier's part number. Additionally, purchasing requirements have been pre-established for specific valve parts based upon a review of the safety function of each part with input from the original manufacturer and/or supplier. These requirements are imposed by the parts purchase order. The PORVs and block valves are included within the scope of the Inservice Testing Program for valves. The block valves are included within the scope of the Motor-Operated Valve Testing Program being developed in response to NRC Generic Letter 89-10.

New Hampshire Yankee submitted a proposed amendment to Seabrook Station Technical Specifications (TS) implementing the guidance of NRC Generic Letter 90-06 by letter number NYN-91167, dated October 16, 1991, for Seabrook Station, Unit 1. In NYN-91167, NHY indicated that the proposed changes regarding PORV operability are based upon the model technical specifications provided as Attachment A-1 of Enclosure A to Generic Letter 90-06. The proposed changes regarding low temperature overpressure protection are based upon the model technical specifications provided as Attachment B-1 to Enclosure B of Generic Letter 90-06, but reflect the fact that either the PORVs or the Residual Heat Removal (RHR) System suction relief valves can provide the required low temperature overpressure protection.

New Hampshire Yankee has developed the Seabrook Station TS changes pertaining to Generic Letter 90-06 in association with the licensees of Braidwood, Byron, Callaway, Comanche Peak, Millstone 3, Vogtle and Wolf Creek. This common approach is facilitated by the similarity of plant types and technical specifications. All the plants in this group are Westinghouse pressurized water reactors (PWR) which utilize the PORVs and RHR System suction relief valves for low temperature overpressure protection.

3.0 EVALUATION FOR GENERIC ISSUE 70 (GI-70)

The actions proposed by the NRC staff to improve the reliability of PORVs and block valves represent a substantial increase in 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 70 are discussed in NUREG-1316, "Technical Findings and Regulatory Analysis Related to Generic Issue 70 - Evaluation of Power-Operated Relief Valve Reliability in PWR Nuclear Power Plants."

3.1 Description of Proposed Changes (GI-70)

The proposed changes to Seabrook Station Technical Specification 3/4.4.4 "Relief Valves" and its associated Bases increases the availability of the PORVs for Reactor Coolant System (RCS) pressure transient mitigation. These changes require that power be maintained to block valves which are closed to isolate PORVs which are exhibiting excessive seat leakage because removal of power would render the block valves inoperable.

By maintaining power to closed block valves when the PORVs are exhibiting excessive seat leakage, the block valves can be readily opened to allow the PORVs to be used to control RCS pressure transients. Closure of block valves establishes RCS pressure boundary integrity for PORVs which are exhibiting excessive seat leakage. If the block valves are inoperable, the changes preclude the potential for having a stuck-open PORV that could not be isolated because of an inoperable block valve, yet maintains the ability to use the PORVs for RCS pressure transient control by placing the associated PORV in manual control (i.e., the control switch in the "CLOSE" position). The proposed changes require that PORVs be cycled only during MODES 3 and 4 and not during power operation to simulate the temperature and environmental effects on the PORVs.

The proposed changes do not reduce the margin of safety defined in its Bases. The changes do not affect the functions of the PORVs and their block valves in MODES 1, 2, or 3. The changes require that if one PORV is inoperable due to causes other than excessive seat leakage, within one hour the PORV must be restored to operable status or the associated block valve must be closed with its power removed. No credit for automatic PORV operation is taken in the FSAR analysis for MODES 1, 2 and 3 transients, and the PORVs can be considered OPERABLE in either the manual or automatic mode. There is no change proposed to the PORV actuation circuitry or to the PORV or block valve power supply configuration. The proposed changes will increase the availability of the PORVs to mitigate RCS pressure transients and will, therefore, enhance safe operation.

4.0 EVALUATION FOR GENERIC ISSUE 94 (GI-94)

The actions proposed by the NRC staff to improve the availability of the low temperature overpressure protection (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 Overpressure Protection for Light-Water Reactors."

4.1 Description of Proposed Changes (GI-94)

The proposed changes to Seabrook Station Technical Specification 3/4.4.9.3 "Overpressure Protection Systems" and its associated Bases provide enhanced operational flexibility through the use of a PORV in combination with an RHR suction relief valve for low temperature overpressurization protection. Each of these relief valves, alone, is capable of mitigating a design basis mass or heat addition transient.

The proposed TS changes require that at least two overpressure protection devices; that is, two PORVs, or two RHR suction relief valves, or one PORV and one RHR suction relief valve, must be OPERABLE when low temperature overpressure protection is required. The NRC found acceptable the use of the RHR suction relief valves for low temperature overpressure protection in NUREG-0896, "Safety Evaluation Report related to the operation of Seabrook Station, Units 1 and 2." Seabrook Station overpressure protection analyses demonstrate that each RHR suction relief valve provides sufficient relief capacity to prevent exceeding 10 CFR 50, Appendix G limits during the overpressurization design bases mass addition event of one charging pump or one safety injection pump operating at full flow with the RCS water solid and loss of letdown capability. The analyses also show that each RHR suction relief valve will prevent exceeding the Appendix G limits during the overpressurization design bases heat addition event of a reactor coolant pump start with the steam generator secondary temperature 50° warmer than RCS temperature.

Seabrook Station Facility Operating License Amendment No. 3, issued July 15, 1991, included the removal of the Residual Heat Removal System suction/isolation valves autoclosure interlock (ACI) function. The RHR ACI removal design change was implemented during the first refueling outage (July 1991 to October 1991). This plant modification enhances RHR system reliability and overpressure protection system availability by precluding spurious RHR suction valve closures caused by potential malfunctions of the ACI circuit. The combination of PORVs and RHR suction relief valves provides an equivalent level of overpressure protection with no degradation in the level of safety. While low temperature overpressure protection is required for all shutdown modes, the most vulnerable period of time was found to be MODE 5 with the reactor coolant temperature less than or equal to 200°F, especially when water solid, based on the detailed evaluation of operating reactor experiences performed in support of Generic Issue 94. The staff concluded that the LTOP systems perform a safety-related function and inoperable overpressure protection equipment should be restored to an operable status in a shorter period of time. The current 7-day allowed outage time (AOT) is considered to be too long under certain conditions and should be reduced to 24 hours when operating in MODE 5 or MODE 6 when the potential for an overpressure transient is most likely to occur. Thus, added assurance of overpressure protection system availability is provided in the TS change by reducing the AOT for one of the two required overpressure protection devices (PORV or RHR suction relief valve) from 7 days to 24 hours in MODE 5 and MODE 6, thereby providing a greater level of safety.

The proposed changes to TS Surveillance Requirements include the performance of an ANALOG CHANNEL OPERATIONAL TEST (ACOT) on the PORV actuation channel to demonstrate that the PORV is OPERABLE when the PORV is being used for low temperature overpressure protection.

The proposed TS changes do not reduce the margin of safety defined in its Bases. The changes do not affect the functions of the PORVs or the Overpressure Protection Systems required in MODE 4 below 329°F, MODE 5 and MODE 6. There is no change to the PORV actuation circuitry or to their PORV power supply configuration and no reduction in surveillance testing of the PORVs or Overpressure Protection Systems. The proposed changes will increase the availability of the PORVs and Overpressure Protection Systems to mitigate RCS pressure transients and will, therefore, enhance safe operation.

5.0 OVERALL PROTECTION

The proposed changes to Seabrook Station Technical Specifications included in New Hampshire Yankee letter number NYN-91167, dated October 16, 1991, are consistent with that proposed in Generic Letter 90-06. The staff has reviewed New Hampshire Yankee's proposed changes to the Seabrook Station Technical Specifications. Since the proposed changes are consistent with the staff's position previously stated in Generic Letter 90-06 and found to be justified in the above mentioned regulatory analysis, the staff finds the proposed amendment to be acceptable.

6.0 STATE CONSULTATION

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

7.0 ENVIRONMENTAL CONSIDERATION

This 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 (57 FR 7815). 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.

8.0 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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