July 15, 1991

Docket No. 50-443

Mr. Ted C. Feigenbaum
President and Chief Executive Officer
New Hampshire Yankee Division
Public Service Company of New Hampshire
Post Office Box 300
Seabrook, New Hampshire 03874

Dear Mr. Feigenbaum:

SUBJECT: ISSUANCE OF AMENDMENT NO. 3 TO FACILITY OPERATING LICENSE NO. NPF-86 - SEABROOK STATION, UNIT NO. 1 (TAC NO. 79624)

The Commission has issued the enclosed Amendment No. 3 to Facility Operating License No. NPF-86 for the Seabrook Station Unit 1. This amendment is in response to your application of January 24, 1991 as supplemented by letter dated May 16, 1991.

This amendment revises the Technical Specifications for Seabrook Station, Unit 1 involving removal of the residual heat removal (RHR) isolation valve autoclosure interlock.

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 Edison

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

Enclosures:
1. Amendment No. 3 to License No. NPF-86
2. Safety Evaluation

cc w/enclosures:
See next page

OFC	:LA:PDI-3	:PM:PDI-3	:OGC	:PD:PDI-3	:
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OFFICIAL RECORD COPY Document Name: SEABROOK AMEND. TAC 79624

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

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G E Edison

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Project Directorate I-3 Division of Reactor
Projects - I/II Office of Nuclear Reactor
Regulation

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2. Safety Evaluation

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

PUBLIC SERVICE COMPANY OF NEW HAMPSHIRE, ET AL*

DOCKET NO. 50-443

SEABROOK STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 3
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 Public Service Company of New Hampshire (the licensee), acting for itself and as agent and representative of the 11 other utilities listed below and hereafter referred to as licensees, dated January 24, 1991 as supplemented by letter dated May 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.

*Public Service Company of New Hampshire is authorized to act as agent for 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.

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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. 3 , and the Environmental Protection Plan contained in Appendix B are incorporated into Facility Operating License No. NPF-86. PSNH 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 prior to restart from the first refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard H. Wessman, 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: July 15, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 3

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 4-34	3/4 4-34
3/4 4-35	3/4 4-35
3/4 5-6	3/4 5-6
B3/4 4-15	B3/4 4-15
B3/4 4-16	B3/4 4-16
--	B3/4 4-17*

*Denotes new page

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.

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following Overpressure Protection Systems shall be OPERABLE:

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

b. 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

c. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 1.58 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. With one PORV and one RHR suction relief valve inoperable, either restore two PORVs or two RHR suction relief valves to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 1.58-square-inch vent within the next 8 hours.

b. With both PORVs and both RHR suction relief valves inoperable, depressurize and vent the RCS through at least a 1.58-square-inch vent within 8 hours.

c. 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.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

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

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 when the PORV is being used for overpressure protection.

4.4.9.3.2 Each RHR suction relief valve shall be demonstrated OPERABLE when the RHR suction relief valves are being used for cold 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 that is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

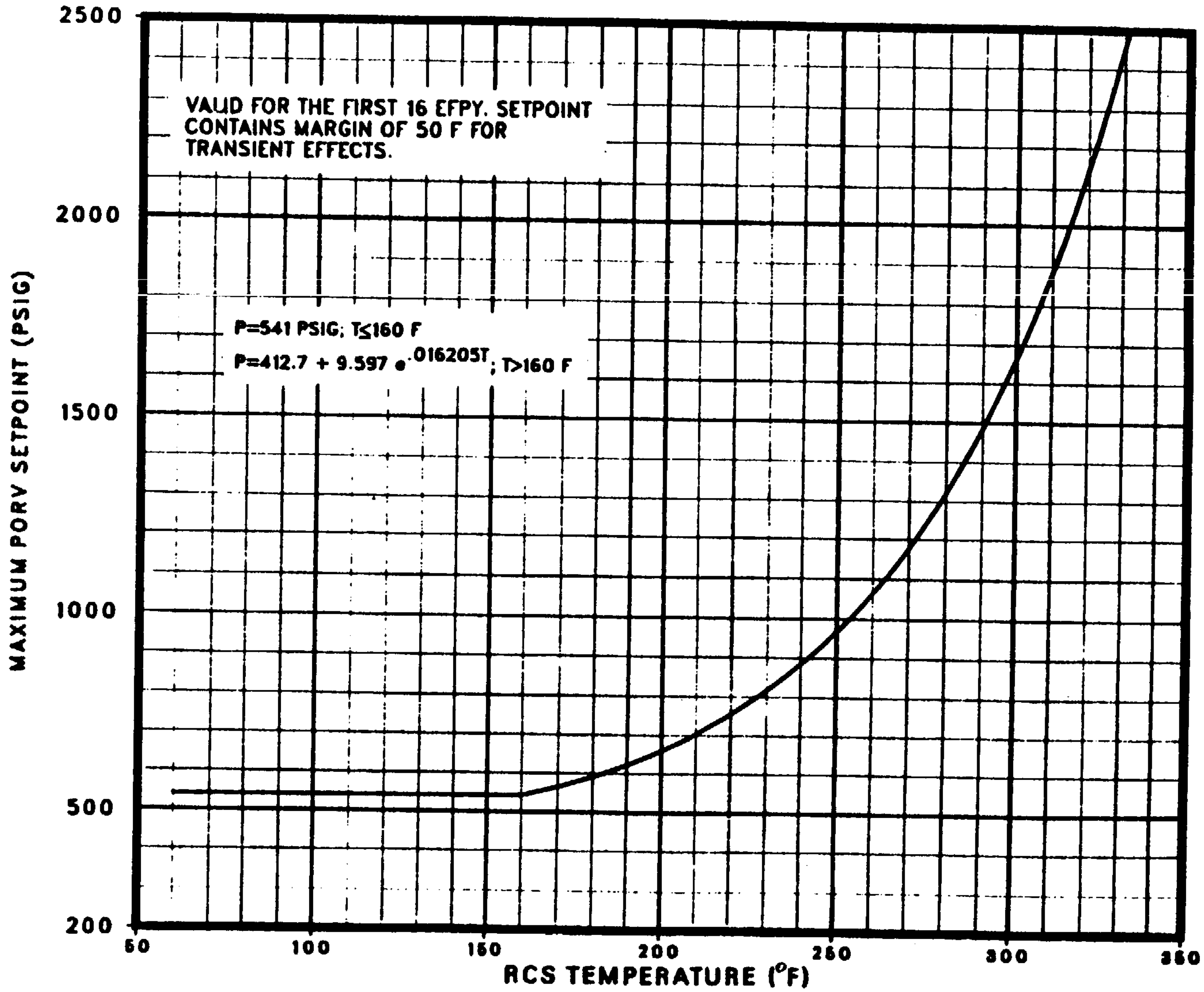

FIGURE 3.4-4 RCS COLD OVERPRESSURE PROTECTION SETPOINTS

ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350°F

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
SI-V-3	Accumulator Isolation	Open*
SI-V-17	Accumulator Isolation	Open*
SI-V-32	Accumulator Isolation	Open*
SI-V-47	Accumulator Isolation	Open*
SI-V-114	SI Pump to Cold-Leg Isolation	Open
RH-V-14	RHR Pump to Cold-Leg Isolation	Open
RH-V-26	RHR Pump to Cold-Leg Isolation	Open
RH-V-32	RHR to Hot-Leg Isolation	Closed
RH-V-70	RHR to Hot-Leg Isolation	Closed
SI-V-77	SI to Hot-Leg Isolation	Closed
SI-V-102	SI to Hot-Leg Isolation	Closed

b. At least once per 31 days by:

1) Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping high points, and

2) Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1) For all accessible areas of the containment prior to establishing PRIMARY CONTAINMENT INTEGRITY, and

2) Of the areas affected within containment at the completion of each containment entry when PRIMARY CONTAINMENT INTEGRITY is established.

*Pressurizer pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS

4.5.2 (Continued)

d. At least once per 18 months by:

   1) Verifying automatic interlock action of the RHR system from the Reactor Coolant System to ensure that with a simulated or actual Reactor Coolant System pressure signal greater than or equal to 365 psig, the interlocks prevent the valves from being opened.

   2) A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

e. At least once per 18 months, during shutdown, by:

   1) Verifying that each automatic valve in the flow path actuates to its correct position on (Safety Injection actuation and Automatic Switchover to Containment Sump) test signals, and

   2) Verifying that each of the following pumps start automatically upon receipt of a Safety Injection actuation test signal:

      a) Centrifugal charging pump,

      b) Safety Injection pump, and

      c) RHR pump.

f. By verifying that each of the following pumps develops the indicated differential pressure on recirculation flow when tested pursuant to Specification 4.0.5:

   1) Centrifugal charging pump, $\geq$ 2480 psid;

   2) Safety Injection pump, $\geq$ 1445 psid; and

   3) RHR pump, $\geq$ 176 psid.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

COLD OVERPRESSURE PROTECTION

The OPERABILITY of two PORVs, or two RHR suction relief valves, 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 Appenix 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. Given the short time duration that this condition is allowed and the low probability of a single failure of a PORV is not assumed. 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.

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.

3/4.4.10 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 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1983 Edition and Addenda through Summer 1983.

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the Reactor Coolant System that could inhibit natural circulation core cooling. The OPERABILITY of least one Reactor Coolant System vent path from the reactor vessel head and the pressurizer steam space ensures that the capability exists to perform this function.

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vents are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plant Requirements," November 1980.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 3 TO FACILITY OPERATING LICENSE NO. NPF-86

PUBLIC SERVICE COMPANY OF NEW HAMPSHIRE

SEABROOK STATION, UNIT NO. 1

DOCKET NO. 50-443

1.0 INTRODUCTION

By letter dated April 22, 1988 (Ref. 1), the Westinghouse Owners' Group (WOG) submitted Topical Report WCAP-11736 entitled "Residual Heat Removal System Autoclosure Interlock Removal Report for the Westinghouse Owners' Group," for NRC review. Westinghouse Report WCAP-11736 documents the analyses performed to justify deletion of the autoclosure interlock (ACI) on the Residual Heat Removal System (RHRS) suction/isolation valves at four reference plants: Salem Unit 1, Callaway Unit 1, North Anna Unit 1, and Shearon Harris Unit 1. The reference plants represent the lead plant in each of four groups into which WOG participating plants were categorized based on similarity of RHRS configuration and design characteristics. The proposed ACI deletion addresses NRC concerns regarding potential failure of ACI circuitry resulting in isolation of the RHRS with attendant loss of decay heat removal capability during cold shutdown and refueling.

A Safety Evaluation Report (SER) documenting the NRC review of WCAP-11736 was issued on August 8, 1989 (Ref. 2). The SER concluded that a net safety benefit would result from removal of the RHRS ACI provided that five plant improvements delineated in the SER are implemented. In addition, the SER concluded that the information contained in WCAP-11736 may be referenced to supplement licensees' plant-specific submittals requesting removal of the RHRS ACI. However, such reference would only be used to show compliance with those items that are generic to the WOG plants. A plant-specific submittal would be required of each licensee seeking approval to remove the RHR ACI.

The above referenced plant improvements are listed below:

(1) An alarm will be added to each RHR suction valve which will actuate if the valve is open and the pressure is greater than the open permissive setpoint and less than the RHR system design pressure minus the RHR pump head pressure.

(2) Valve position indication to the alarm must be provided from stem-mounted limit switches and power to these switches must not be affected by power lockout of the valve.

(3) The procedural improvements described in WCAP-11736 should be implemented. Procedures themselves are plant-specific.

(4) Where feasible, power should be removed from the RHR suction valves prior to their being leak-checked (plant-specific).

(5) The RHR suction valve operators should be sized so that the valves cannot be opened against full system pressure (plant-specific).

2.0 EVALUATION

By letter dated January 24, 1991, New Hampshire Yankee, licensee for the Seabrook Station, submitted an application to revise Technical Specifications (TS) 4.5.2.d.1, 3.4.9.3.a, and 4.4.9.3.2 (a and b) (Ref. 3). Supplementary information was provided by letter dated May 16, 1991 (Ref. 4). These TS revisions have been proposed in support of the licensee's plans to remove the RHRS ACI during their 1991 refueling outage. The proposed revision to TS 4.4.9.3.2 deletes the surveillance requirement to verify once every 31 days that one of the two in-series suction valves in each RHRS train is in the open position with its power removed, and to verify once every 12 hours that the second suction valve in each train is open. This is replaced by the requirement to verify once every 72 hours that both suction valves in each train are open. The TS applies when the RHR suction relief valves are being used for cold overpressure protection. The proposed revision to TS 3.4.9.3.a reduces the RHR suction relief valve setpoint upper limit from 450 psig ± 3 percent to 450 psig + 0, - 3 percent. Regarding TS 4.5.2.d.1, the proposed revision deletes the surveillance requirement for verifying ACI operability (the open permissive interlock surveillance remains unchanged).

As noted above, the NRC-approved Westinghouse report WCAP-11736 provides the underlying basis for justifying the licensee's planned action. The WCAP-11736 reference plant for Seabrook is Callaway Unit 1. The licensee's submittals (Refs. 3 and 4) include a plant-specific analysis of the planned ACI deletion as a supplement to WCAP-11736. The submittal includes a delineation of the relevant design/operational differences that exist between Seabrook and the reference plant as described in WCAP-11736. The licensee has examined these differences to determine their impact on ISLOCA potential, RHRS availability, low-temperature overpressure protection, and on the conclusions reached in WCAP-11736. In addition, the licensee has addressed each of the five plant improvements set forth in Reference 2 and listed above. Where deviations from these improvements are proposed by the licensee, analyses are presented to demonstrate that equivalent levels of safety exist.

With regard to the above mentioned five plant improvements, the licensee's January 24 and May 16, 1991 submittals have provided the following responses:

° Concerning Improvement 1, the existing Seabrook design already incorporates an alarm for each RHRS suction valve which will activate if the valve is not fully closed when RCS pressure exceeds the alarm setpoint. The setpoint (365 psig) is consistent with the WCAP-11736 guidance. Also, in accordance with WCAP-11736, the open permissive interlock (OPI) for each RHRS suction valve will remain intact and unchanged.

- Concerning Improvement 2, the Seabrook design utilizes existing motor operator limit switch contacts for valve position input to the existing alarms. These contacts are different from the limit switch contacts which provide valve position on the main control board. Furthermore, valve position alarms remain operational during valve power lockout. The original intent of using stem-mounted limit switches in the alarm circuit was to provide a diverse means of valve position indication. Since the existing design already provides this diversity, the licensee does not plan to install stem-mounted limit switches.

- Concerning Improvement 3, the licensee has reviewed the Seabrook operating procedures to determine the effect of ACI removal and has committed to make the appropriate revisions. The procedures reviewed include those delineated in Reference 2. If an open RHR suction valve cannot be closed upon receipt of an alarm, operators will be directed to halt RCS pressurization and the plant will be returned to the shutdown cooling mode. To further ensure alarm operability, instrument loop calibration procedures will be revised.

- Concerning Improvement 4, the licensee ~~does not~~ plan to remove power from the RHRS suction/isolation valves prior to leak testing. The original intent of this recommended improvement was to ensure that the valves remained in the tested configuration during testing. Leak testing of the RHRS suction/isolation valves at Seabrook is normally performed in Mode 4, 5, or 6. Closure and power removal from these valves is required only prior to entering Mode 3. One advantage of performing leak testing prior to entry into Mode 3 is that the amount of cooldown required to perform valve maintenance in the event the valve exhibits greater than allowable leakage is minimized. The two RHR trains are leak tested consecutively. When testing is completed on one train, that train is returned to service. Removal of valve power prior to testing in Modes 4, 5, and 6 (and subsequent restoration of power) would increase procedural complexity and time without offering a safety benefit. Increased testing time would decrease the availability of the RHRS to remove decay heat and for the RHRS suction relief valves to assist in low-temperature overpressure protection. It also should be noted that, since the operable loop as well as the inoperable loop would be subjected to any unlikely pressure transients occurring during Modes 4, 5, or 6, removal of power to the suction valve being tested (on the inoperable loop) would not alter the impact of a pressure transient in RHRS piping inside or outside of containment.

- Concerning Improvement 5, the licensee has stated that the RHRS suction valves potentially have the capability of opening against full RCS pressure. However, these valves are provided with an OPI feature which prevents opening when RCS pressures exceed 365 psig. The OPI is tested in accordance with TS 4.5.2.d.1 once every 18 months. Additionally, these valves are de-energized during power operation. Therefore, the likelihood of an ISLOCA scenario owing to an inadvertent open signal when the RCS is at full pressure is extremely low. On this basis, the licensee does not plan to downsize the motor actuators.

We have completed our evaluation of the licensees January 24 and May 16, 1991 submittals and have concluded the following:

- The licensee has adequately identified differences in RHRS configuration and design/operational characteristics that exist between Seabrook and the reference plant (Callaway) addressed in WCAP-11736. Because these differences are insignificant, the analyses and conclusions presented in WCAP-11736 for Callaway are directly applicable to Seabrook.

- The licensee has adequately addressed the five plant improvements delineated in Reference 2. Where deviations between these improvements and the licensee's proposed actions were identified, the licensee has adequately demonstrated that the proposed actions provided at least an equivalent level of safety.

- The proposed change to TS 4.5.2.d.1 (i.e., to delete the surveillance requirement for verifying ACI operability) is consistent with the licensee's plans to remove the ACI feature from the RHRS suction valves. This change is, therefore, acceptable.

- The proposed change to TS 3.4.9.3.a (i.e., to reduce the RHRS suction relief valve setpoint upper limit from 450 psig ± 3 percent to 450 psig + 0, - 3 percent) provides additional margin for overpressure protection and thus represents a change in a conservative direction. This revision is, therefore, acceptable.

- The proposed change to TS 4.4.9.3.2 (a and b) includes several revisions (see earlier description). The current TS requirement to periodically verify that power is removed from one of the two in-series RHRS suction/isolation valves in each train is intended to ensure that a single failure of either of the two common pressure transmitters (which provide the ACI signal to these valves) does not result in both RHRS trains becoming isolated from the RCS. With the planned removal of the ACI circuitry, however, the only mechanism that can cause an isolation of both RHRS trains is now eliminated. Therefore, valve power removal and its associated surveillance requirement become unnecessary. The revised TS requires only that these valves be verified open at least once every 72 hours. Additionally, for the second of the two in-series isolation valves in each RHRS train, verification of the open position is changed from once every 12 hours to once every 72 hours. The 72 hour surveillance frequency for the isolation valves is consistent with that specified for the reference plant (Callaway) in the already approved WCAP-11736. This frequency now becomes identical to the existing surveillance frequency for verifying the open position of the power operated relief valves (PORVs).

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

This 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. 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 exposure. The Commission has previously published a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (56 FR 27048). Accordingly, this 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 this amendment.

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

6.0 REFERENCES

1. R. A. Newton, Chairman WOG, letter to NRC, dated April 22, 1988.

2. A. Thadani (NRC) letter to R. A. Newton, Chairman WOG, "Acceptance for Reference WCAP-11736, Rev. 0, 'Residual Heat Removal System, Auto Closure Interlock (ACI) Removal Report' in Plant Specific Submittals," dated August 8, 1989.

3. T. C. Feigenbaum (NHY), letter to NRC, dated January 24, 1991.

4. T. C. Feigenbaum (NHY), letter to NRC, dated May 16, 1991.

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