50-443

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

March 12, 1997

Mr. Ted C. Feigenbaum
Executive Vice President and
Chief Nuclear Officer
Northeast Utilities Service Company
c/o Mr. Terry L. Harpster
Director - Nuclear Licensing Services
P.O. Box 128
Waterford, CT 06385

SUBJECT: AMENDMENT NO. 50 TO FACILITY OPERATING LICENSE NPF-86: RELOCATION OF CERTAIN REQUIREMENTS TO SEABROOK STATION TECHNICAL REQUIREMENTS MANUAL - LICENSE AMENDMENT REQUEST 96-02 (TAC NO. M96723)

Dear Mr. Feigenbaum:

The Commission has issued the enclosed Amendment No. 50 to Facility Operating License No. NPF-86 for the Seabrook Station, Unit No. 1, in response to your application dated October 17, 1996.

The amendment revises the Appendix A Technical Specifications relating to the in-core detector system, seismic instrumentation, meteorological instrumentation, and turbine overspeed protection. Specifically, the amendment authorizes the relocation of Limiting Conditions for Operation 3.3.3.2, 3.3.3.3, 3.3.3.4, and 3.3.4 and Surveillance Requirements 4.3.3.3.1, 4.3.3.3.2, 4.3.3.4, 4.3.4.1, and 4.3.4.2. The relocated requirements are to be incorporated into the Seabrook Station Technical Requirements Manual (SSTR) as described in your October 17, 1996, application and as evaluated in the enclosed Safety Evaluation. Technical Specification 5.5 is deleted but will not be relocated to the SSTR.

As administrative actions by the Commission involving only the format of the license, the Commission has amended the Seabrook Operating License to redesignate Paragraph 2.(J) as Paragraph 3., and has added a new Paragraph 2.(J), "Additional Conditions," to document the North Atlantic commitment to relocate the above mentioned technical specification requirements to the SSTR. The Commission also has amended the license to include a new Appendix C which provides a listing of additional license conditions beginning with this license condition. These administrative actions do not authorize any activities outside the scope of your application. The inclusion of this license condition in the license has been discussed with your staff on March 6, 1997, and your staff has agreed to the license condition.

DFO1

CP-1

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

Sincerely,

(Original Signed By)

Albert W. De Agazio, Sr. Project Manager
Project Directorate I-3
Division of Reactor Projects -I/II
Office of Nuclear Reactor Regulation

Docket No. 50-443
Serial No. SEA-96-004

Enclosures: 1. Amendment No. 50 to NPF-86
2. Safety Evaluation

cc w/encls: See next page

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

Sincerely,

Albert W. De Agazio

Albert W. De Agazio, Sr. Project Manager
Project Directorate I-3
Division of Reactor Projects -I/II
Office of Nuclear Reactor Regulation

Docket No. 50-443
Serial No. SEA-96-004

Enclosures: 1. Amendment No. 50 to NPF-86
2. Safety Evaluation

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cc:

Lillian M. Cuoco, Esq.
Senior Nuclear Counsel
Northeast Utilities Service Company
P.O. Box 270
Hartford, CT 06141-0270

Mr. Peter Brann
Assistant Attorney General
State House, Station #6
Augusta, ME 04333

Resident Inspector
U.S. Nuclear Regulatory Commission
Seabrook Nuclear Power Station
P.O. Box 1149
Seabrook, NH 03874

Jane Spector
Federal Energy Regulatory Commission
825 North Capital Street, N.E.
Room 8105
Washington, DC 20426

Town of Exeter
10 Front Street
Exeter, NH 03823

Mr. George L. Iverson, Director
New Hampshire Office of Emergency Management
State Office Park South
107 Pleasant Street
Concord, NH 03301

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Office of the Attorney General
One Ashburton Place
20th Floor
Boston, MA 02108

Board of Selectmen
Town of Amesbury
Town Hall
Amesbury, MA 01913

Mr. Jack Dolan
Federal Emergency Management Agency
Region I
J.W. McCormack P.O. &
Courthouse Building, Room 442
Boston, MA 02109

Mr. David Rodham, Director
ATTN: James Muckerheide
Massachusetts Civil Defense Agency
400 Worcester Road
P.O. Box 1496
Framingham, MA 01701-0317

Jeffrey Howard, Attorney General
G. Dana Bisbee, Deputy Attorney General
33 Capitol Street
Concord, NH 03301

Mr. D. M. Goebel
Vice President-Nuclear Oversight
Northeast Utilities Service Company
P. O. Box 270
Hartford, CT 06141-0270

Mr. J. K. Thayer
Recovery Officer, Nuclear Engineering and Support
Northeast Utilities Service Company
P.O. Box 128
Waterford, CT 06385

Mr. F. C. Rothen
Vice President - Nuclear Work Services
Northeast Utilities Service Company
P.O. Box 128
Waterford, CT 06385

Mr. A. M. Callendrello
Licensing Manager - Seabrook Station
North Atlantic Energy Service Corp.
P.O. Box 300
Seabrook, NH 03874

cc:

Mr. W. A. DiProfio
Nuclear Unit Director
Seabrook Station
North Atlantic Energy Service Corporation
P.O. Box 300
Seabrook, NH 03874

Mr. Frank W. Getman, Jr.
Cocheco Falls Millworks
100 Main Street, Suite 201
Dover, NH 03820

Mr. B. D. Kenyon
President - Nuclear Group
Northeast Utilities Service Group
P.O. Box 128
Waterford, CT 06385

Mr. B. L. Drawbridge
Executive Director Services &
Senior Site Officer
North Atlantic Energy Service Corp.
Seabrook, NH 03874

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

NORTH ATLANTIC ENERGY SERVICE CORPORATION, ET AL*

DOCKET NO. 50-443

SEABROOK STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 50
License No. NPF-86

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

   A. The application for amendment by North Atlantic Energy Service Corporation, et al. (the licensee), dated October 17, 1996, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this 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.

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*North Atlantic Energy Service Company (NAESCO) is authorized to act as agent for the: North Atlantic Energy Corporation, Canal Electric Company, The Connecticut Light and Power Company, Great Bay Power Corporation, Hudson Light and Power Department, Massachusetts Municipal Wholesale Electric Company, Montaup Electric Company, New England Power Company, New Hampshire Electric Cooperative, Inc., Taunton Municipal Light Plant, and The United Illuminating Company, 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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 50, 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.

   In addition, the license is amended by redesignating existing paragraph 2.(J) as paragraph 3. and adding new paragraph 2.(J) to read as follows:

   (J) <u>Additional Conditions</u>

   The Additional Conditions contained in Appendix C as revised through Amendment No. 50, are hereby incorporated into this license. NAESCO shall operate the facility in accordance with the Additional Conditions.

   3. This license is effective as of the date of issuance and shall expire at midnight on October 17, 2026.

3. This license amendment is effective as of the date of its issuance, to be implemented within 60 days of issuance. Implementation of this amendment shall include the relocation of certain technical specification requirements to the appropriate licensee-controlled documents as described in the Licensee's application dated October 17, 1996, and evaluated in the staff's Safety Evaluation attached to this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

David S. [illegible] for H. B. Barkow

Patrick A. Milano, Acting Director
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: 1. Page 6 of the license and page 1 of Appendix C to the license
2. Changes to the Technical Specifications

Date of Issuance: March 12, 1997

J. Additional Conditions

The Additional Conditions contained in Appendix C, as revised through Amendment No. 50, are hereby incorporated into this license. NAESCO shall operate the facility in accordance with the Additional Conditions.

3. This license is effective as of the date of issuance and shall expire at midnight on October 17, 2026.

FOR THE NUCLEAR REGULATORY COMMISSION

(Original signed by:
Thomas E. Murley)

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Attachments/Appendices:
1. Appendix A - Technical Specifications (NUREG-1386)
2. Appendix B - Environmental Protection Plan
3. Appendix C - Additional Conditions

Date of Issuance: March 15, 1990

Revised: March 12, 1997

APPENDIX C

ADDITIONAL CONDITIONS
OPERATING LICENSE NO. NPF-86

North Atlantic Energy Service Company (NAESCO) shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
50	NAESCO is authorized to relocate certain technical specification requirements to licensee-controlled documents. Implementation of this amendment shall include the relocation of these technical specification requirements to the appropriate documents, as described in the licensee's application dated October 17, 1996, and evaluated in the staff's Safety Evaluation attached to this amendment.	The amendment shall be implemented within 60 days from March 12, 1997.

ATTACHMENT TO LICENSE AMENDMENT NO. 50

FACILITY OPERATING LICENSE NO. NPF-86

DOCKET NO. 50-443

Replace the following pages of Appendix A, Technical Specifications, with the attached pages as indicated. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change. Overleaf pages have been provided.*

Remove	Insert
iii*	iii*
iv	iv
v	v
vi*	vi*
ix*	ix*
x	x
xiii	xiii
xiv*	xiv*
3/4 3-39*	3/4 3-39*
3/4 3-40	3/4 3-40
3/4 3-41	3/4 3-41
3/4 3-42	3/4 3-42
3/4 3-43	3/4 3-43
3/4 3-44	3/4 3-44
3/4 3-45	3/4 3-45
3/4 3-46*	3/3 3-46*
3/4 3-67	3/4 3-67
B 3/4 3-3*	B 3/4 3-3*
B 3/4 3-4	B 3/4 3-4
B 3/4 3-5*	B 3/4 3-5*
B 3/4 3-6	B 3/4 3-6
5-9	5-9
5-10*	5-10*

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION — PAGE

3.0/4.0 BASES

5.0 DESIGN FEATURES

6.0 ADMINISTRATIVE CONTROLS

6.0 ADMINISTRATIVE CONTROLS

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	DIGITAL CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Containment				
a. Containment - Post LOCA - Area Monitor	S	R	Q	All
b. RCS Leakage Detection				
1) Particulate Radio-activity	S	R	Q	1, 2, 3, 4
2) Gaseous Radioactivity	S	R	Q	1, 2, 3, 4
2. Containment Ventilation Isolation				
a. On Line Purge Monitor	S	R	Q	1, 2, 3, 4
b. Manipulator Crane Area Monitor	S	R	Q	6#
3. Main Steam Line	S	R	Q	1, 2, 3, 4
4. Fuel Storage Pool Areas				
a. Radioactivity-High-Gaseous Radioactivity	S	R	Q	*
5. Control Room Isolation				
a. Air Intake Radiation Level				
1) East Air Intake	S	R	Q	All
2) West Air Intake	S	R	Q	All
6. Primary Component Cooling Water				
a. Loop A	S	R	Q	All
b. Loop B	S	R	Q	All

TABLE NOTATIONS

* With irradiated fuel in the fuel storage pool areas.

\# During CORE ALTERNATIONS or movement of irradiated fuel within the containment.

3.3.3.2 (THIS SPECIFICATION NUMBER IS NOT USED)

3.3.3.3 (THIS SPECIFICATION NUMBER IS NOT USED

TABLE 3.3-7

(THIS TABLE NUMBER IS NOT USED)

<u>TABLE 4.3-4</u>

(THIS TABLE NUMBER IS NOT USED)

3.3.3.4 (THIS SPECIFICATION NUMBER IS NOT USED)

<u>TABLE 3.3-8</u>

(THIS TABLE NUMBER IS NOT USED)

REMOTE SHUTDOWN SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.5 The Remote Shutdown System transfer switches, power, controls and monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE remote shutdown monitoring channels less than the Minimum Channels OPERABLE as required by Table 3.3-9, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours. The provisions of Specification 3.0.4 are not applicable.

b. With the number of OPERABLE remote shutdown monitoring channels less than the Total Number of Channels as required by Table 3.3-9, within 60 days restore the inoperable channel(s) to OPERABLE status or, pursuant to Specification 6.8.2, submit a Special Report that defines the corrective action to be taken.

c. With one or more Remote Shutdown System transfer switches, power, or control circuits inoperable, restore the inoperable switch(s)/circuit(s) to OPERABLE status within 7 days, or be in HOT STANDBY within the next 12 hours. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.5.1 Each remote shutdown monitoring instrumentation channel in Table 3.3-9 shall be demonstrated OPERABLE:

a. Every 31 days by performance of a CHANNEL CHECK, and

b. Every 18 months by performance of a CHANNEL CALIBRATION.

4.3.3.5.2 Each Remote Shutdown System transfer switch, power and control circuit listed in Table 3.3-9, including the actuated components, shall be demonstrated OPERABLE at least once per 18 months.

3/4.3.4 (THIS SPECIFICATION NUMBER IS NOT USED)

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

Injection pumps start and automatic valves position, (2) Reactor trip, (3) feedwater isolation, (4) startup of the emergency diesel generators, (5) containment spray pumps start and automatic valves position, (6) containment isolation, (7) steam line isolation, (8) turbine trip, (9) emergency feedwater pumps start and automatic valves position, (10) containment cooling fans start and automatic valves position, and (11) automatic service water valves position.

The Engineered Safety Features Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates Turbine trip, closes main feedwater valves on T_{avg} below Setpoint, prevents the opening of the main feedwater valves which were closed by a Safety Injection or High Steam Generator Water Level signal, allows Safety Injection block so that components can be reset or tripped.

Reactor not tripped - prevents manual block of Safety Injection.

P-11 On increasing pressurizer pressure, P-11 automatically reinstates Safety Injection actuation on low pressurizer pressure. On decreasing pressure, P-11 allows the manual block of Safety Injection actuation on low pressurizer pressure, and the manual block of SI and steamline isolation on steamline low pressure. On the manual block of steamline low pressure, manual block of steamline low pressure automatically initiates steamline isolation on steam generator pressure negative rate - high.

P-14 On increasing steam generator water level, P-14 automatically trips the turbine and all feedwater isolation valves; inhibits feedwater control valve modulation; and blocks the start of the startup feedwater pump.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that: (1) the associated action will be initiated when the radiation level monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, and (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance. The radiation monitors for plant operations sense radiation levels in selected plant systems and locations and determine whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents

MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS (Continued)

and abnormal conditions. Once the required logic combination is completed, the system sends actuation signals to initiate alarms or automatic isolation action and actuation of Emergency Exhaust or Ventilation Systems.

3/4.3.3.2 (THIS SPECIFICATION NUMBER IS NOT USED)

3/4.3.3.3 (THIS SPECIFICATION NUMBER IS NOT USED)

3/4.3.3.4 (THIS SPECIFICATION NUMBER IS NOT USED)

3/4.3.3.5 REMOTE SHUTDOWN SYSTEM

The OPERABILITY of the Remote Shutdown System ensures that sufficient capability is available to permit safe shutdown of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of Appendix A to 10 CFR Part 50.

MONITORING INSTRUMENTATION

3/4.3.3.5 REMOTE SHUTDOWN SYSTEM (Continued)

The OPERABILITY of the Remote Shutdown System ensures that a fire will not preclude achieving safe shutdown. The Remote Shutdown System instrumentation, control, and power circuits and transfer switches necessary to eliminate effects of the fire and allow operation of instrumentation, control and power circuits required to achieve and maintain a safe shutdown condition are independent of areas where a fire could damage systems normally used to shut down the reactor. This capability is consistent with General Design Criterion 3 and Appendix R to 10 CFR Part 50.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," May 1983 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.3.3.7 (This specification number is not used.)

3/4.3.3.8 (This specification number is not used.)

3/4.3.3.9 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.10 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the WASTE GAS HOLDUP SYSTEM. The OPERABILITY and use of this instrumentation is consistent with the requirements

MONITORING INSTRUMENTATION

3/4.3.3.10 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION (Continued)

of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50. The sensitivity of any noble gas activity monitors used to show compliance with the gaseous effluent release requirements of Specification 3.11.2.2 shall be such that concentrations as low as 1 X 10^{-6} μci/cc are measurable.

3/4.3.4 (THIS SPECIFICATION NUMBER IS NOT USED)

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The containment building is designed and shall be maintained for a maximum internal pressure of 52.0 psig and a temperature of 296°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 193 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 144 inches. The initial core loading shall have a maximum enrichment of 3.15 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 5.0 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 57 full-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80% silver, 15% indium, and 5% cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 12,255 cubic feet at a nominal T_{avg} of 588.5°F.

5.5 (THIS SPECIFICATION NUMBER IS NOT USED)

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes margin for uncertainty in calculation methods and mechanical tolerances with a 95% probability at a 95% confidence level.

b. A nominal 10.35 inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The new fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes margin for uncertainty in calculational methods and mechanical tolerances with a 95% probability at a 95% confidence level.

b. A k_{eff} equivalent to less than or equal to 0.98 when aqueous foam moderation is assumed, which includes margin for uncertainty in calculational methods and mechanical tolerances with a 95% probability at a 95% confidence level.

c. A nominal 21 inch center-to-center distance between fuel assemblies placed in the storage racks.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 14 feet 6 inches.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1236 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 50 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 application dated October 17, 1996, North Atlantic Energy Service Corporation (North Atlantic/the licensee) proposed an amendment to the Appendix A Technical Specifications (TSs) for the Seabrook Station, Unit 1 (Seabrook). The proposed amendment would relocate certain instrumentation requirements stated in TS 3/4.3, *Instrumentation* in accordance with the guidance in Generic Letter 95-10, "Relocation of Selected Technical Specifications Requirements Related to Instrumentation." North Atlantic has committed to relocate the deleted requirements to the Seabrook Station Technical Requirements Manual (SSTR) which is incorporated into the FSAR such that future changes could be made under 10 CFR 50.59. The associated bases for the deleted TS requirements would be deleted also, but they would not be incorporated into the SSTR. The following Limiting Conditions for Operation (LCOs) and associated Surveillance Requirements (SRs) would be relocated to the SSTR:

Technical Specification	Title
LCO - 3.3.3.2	Incore Detector System
LCO - 3.3.3.3 and associated SRs & Tables	Seismic Instrumentation
LCO - 3.3.3.4 and associated SRs & Tables	Meteorological Instrumentation
LCO - 3.3.4 and associated SRs	Turbine Overspeed Protection

The proposed amendment also would delete (without relocating to the SSTR) the reference to the location of the meteorological tower from Technical Specification 5.5.

2.0 BACKGROUND

Section 182a of the Atomic Energy Act of 1954, as amended (the Act) requires applicants for nuclear power plant operating licenses to include technical specifications as part of the license. The Commission's regulatory requirements related to the content of technical specifications are set forth in 10 CFR 50.36. That regulation requires that the technical specifications include items in five specific categories, including (1) safety limits, limiting safety system settings and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. The regulation does not specify the particular requirements to be included in the technical specifications.

The Commission, however, provided guidance for technical specification contents in its "Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," which was published in the Federal Register at 58 FR 39132 (July 22, 1993). The Commission indicated therein that compliance with its Final Policy Statement satisfies Section 182a of the Act. Criteria for the content of technical specifications were subsequently incorporated into 10 CFR 50.36, cf. 60 FR 36953 (July 19, 1995). In particular, the Commission indicated that certain items could be relocated from the technical specifications to licensee-controlled documents, consistent with the standard enunciated in *Portland General Electric Co.* (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979). In that case, the Atomic Safety and Licensing Appeal Board indicated that "technical specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety."

The four criteria defined by 10 CFR 50.36 for determining whether a particular matter is required to be included in the technical specification limiting conditions for operations, are as follows:

(1) Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary;

(2) a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;

(3) a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;

(4) a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Existing technical specification requirements which fall within or satisfy any of the above criteria must be retained in the Technical Specifications; those requirements which do not fall within or satisfy these criteria may be relocated to other licensee-controlled documents.

3.0 EVALUATION

3.1 In-core detector system

North Atlantic has proposed to remove LCO 3.3.3.2, "In-core Detector System", from the Technical Specifications and relocate corresponding requirements for the system in the SSTR. In-core instrumentation is used periodically to calculate reactor core power peaking factors to verify nuclear design predictions, ensure operation within established fuel performance limits, and calibrate other nuclear instrumentation. The measurements are used in a confirmatory manner and do not provide direct input to reactor protection system or engineered safety features actuation system functions.

These instruments are not used for and are not capable of detecting a significant abnormal degradation of the reactor coolant pressure boundary before a design basis accident. These instruments do not function as a primary success path to mitigate events which assume a failure of or a challenge to the integrity of fission product barriers. Core power distributions (measured by the in-core detectors) constitute an important initial condition to design basis accidents and therefore need to be addressed by technical specifications. However, the detectors themselves are not an active design feature needed to preclude analyzed accidents or transients. Therefore, the staff finds that the in-core detector requirements do not meet the criteria of 10 CFR 50.36 for inclusion in technical specifications. Therefore, removal of the in-core instrumentation requirements from the Technical Specifications and relocation of corresponding requirements to the SSTR is acceptable. Any subsequent changes to the provisions may be controlled pursuant to 10 CFR 50.59.

3.2 Seismic Instrumentation

North Atlantic has proposed to remove LCO 3.3.3.3, "Seismic Instrumentation", and associated SRs and tables from the Technical Specifications; corresponding requirements would be relocated in the SSTR. Section VI(a)(3) of Appendix A to 10 CFR Part 100 requires that seismic monitoring instrumentation be provided to determine promptly the response of those nuclear power plant features important to safety in the event of an earthquake. This capability is required to allow for a comparison of the measured response to that used in the design basis for the unit. Comparison of such data is needed to (1) determine whether the plant can continue to be operated safely and (2) permit such timely action as may be appropriate. However, the seismic instrumentation does not actuate any protective equipment or serve any direct role in the mitigation of an accident.

The capability of the plant to withstand a seismic event or other design basis accident is determined by the initial design and construction of systems, structures, and components. The instrumentation is used to alert operators to the seismic event and evaluate the plant response. The Final Policy Statement explained that instrumentation to detect precursors to reactor coolant pressure boundary leakage, such as seismic instrumentation, is not included in the first criterion. As discussed above, the seismic instrumentation is not a protective design feature or part of a primary success path for events that challenge fission product barriers. The staff has concluded that the seismic monitoring instrumentation does not satisfy the criteria stated in 10 CFR 50.36. Therefore, removal of seismic monitoring instrumentation requirements from the Technical Specifications and relocation of corresponding requirements to the SSTR is acceptable. Any subsequent changes to the provisions may be controlled pursuant to 10 CFR 50.59.

3.3 Meteorological Instrumentation

North Atlantic has proposed to remove LCO 3.3.3.4, "Meteorological Instrumentation," and associated SRs and tables from the Technical Specifications; corresponding requirements would be relocated in the SSTR. In 10 CFR 50.47, "Emergency Plans," and 10 CFR Part 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," the Commission requires power plant licensees to provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. Timely access to accurate local meteorological data is important for estimating potential radiation doses to the public and for determining appropriate protective measures.

In 10 CFR 50.36a(a)(2), the Commission requires nuclear power plant licensees to submit annual reports specifying the quantity of each of the principal radionuclides released to unrestricted areas in liquid and airborne effluents and such other information as may be required by the NRC to estimate maximum potential annual radiation doses to the public. A knowledge of meteorological conditions in the vicinity of the reactor is important in providing a basis for estimating annual radiation doses resulting from radioactive materials released in airborne effluents.

Accordingly, the meteorological instrumentation serves a useful function in estimating radiation doses to the public from either routine or accidental releases of radioactive materials to the atmosphere.

The meteorological instrumentation does not serve a primary protective function so as to warrant inclusion in the technical specifications in accordance with the 10 CFR 50.36 criteria. The instrumentation does not serve to ensure that the plant is operated within the bounds of initial conditions assumed in design basis accident and transient analyses or that the plant will be operated to preclude transients or accidents. The meteorological instrumentation does not serve as part of the primary success path of a safety sequence analysis used to demonstrate that the consequences of these events are within the appropriate acceptance criteria. Accordingly, the staff has concluded that the meteorological instrumentation does not meet the 10 CFR 50.36 criteria and need not be included in technical specifications.

Therefore, removal of the meteorological instrumentation requirements from the Technical Specifications and relocation of corresponding requirements to the SSTR is acceptable. Any subsequent changes to the provisions may be controlled pursuant to 10 CFR 50.59.

3.4 Turbine Overspeed Protection

North Atlantic has proposed to remove LCO 3.3.4, "Turbine Overspeed Protection," and associated SRs from the Technical Specifications; corresponding requirements would be relocated in the SSTR. The turbine is equipped with control valves and stop valves which control turbine speed during normal plant operation and protect it from overspeed during abnormal conditions. The turbine overspeed protection system consists of separate mechanical and electrical sensing mechanisms which are capable of initiating fast closure of the control and stop valves.

General Design Criterion 4 of Appendix A to 10 CFR Part 50 requires that structures, systems, and components important to safety be appropriately protected from the effects of missiles that may result from equipment failures. Application of the design criteria to turbine missiles is described in SRP Section 10.2 and in subsequent safety evaluations related to probabilities of turbine failures, turbine orientations, and surveillance requirements for turbine overspeed protection systems. In NUREG-1366, "Improvements to Technical Specification Surveillance Requirements," the staff discusses the benefits, resultant costs, and the safety impact of performing turbine overspeed protection surveillances.

Although the design basis accidents and transients include a variety of system failures and conditions which might result from turbine overspeed events and potential missiles striking various plant systems and equipment, the system failures and plant conditions are much more likely to be caused by events other than turbine failures. In view of the low likelihood of turbine missiles, assumptions related to the turbine overspeed protection system are not part of an initial condition of a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The turbine overspeed protection system is not relied upon in the design basis accident or transient analyses as a primary success path to mitigate such events.

Probabilistic safety assessments and operating experience have demonstrated that proper maintenance of the turbine overspeed control valves is important to minimize the potential for overspeed events and turbine damage; however, that experience has also demonstrated that there is low likelihood of significant risk to public health and safety because of turbine overspeed events. Further, the potential for and consequences of turbine overspeed events are diminished by factors such as the orientation of the turbine relative to plant structures and equipment, licensee inservice testing programs, which must comply with 10 CFR 50.55(a), and surveillance programs for the turbine control and stop valves derived from the manufacturer's recommendations.

Accordingly, the staff finds that the turbine overspeed protection system does not meet the 10 CFR 50.36 criteria. Therefore, removal of the turbine overspeed protection instrumentation requirements from the Technical Specifications and relocation of corresponding requirements to the SSTR is acceptable. Any subsequent changes to the provisions may be controlled pursuant to 10 CFR 50.59.

3.5 Implementation

In its application, North Atlantic committed to insert the relocated requirements in the SSTR, but did not indicate the date by which this will be accomplished. North Atlantic's commitment is incorporated in new paragraph 2.(J), which incorporates new Appendix C to the license. North Atlantic's commitment to relocate these items to the SSTR is required to be accomplished within 60 days of issuance of this amendment, as set forth in Appendix C to the license.

4.0 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.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes a surveillance requirement. 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 (61 FR 66713). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Albert W. De Agazio

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