Docket No. 50-443
Serial No. SEA-93-004

April 7, 1993

Docket File

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: AMENDMENT NO. 19 TO FACILITY OPERATING LICENSE NPF-86: REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITS - LICENSE AMENDMENT REQUEST 92-06 (TAC M84344)

The Commission has issued the enclosed Amendment No. 19 to Facility Operating License No. NPF-86 for the Seabrook Station, Unit No. 1, in response to your application dated August 17, 1992 (NYN-92111).

The amendment revises the Appendix A Technical Specifications (TSs) relating to Reactor Coolant System heatup and cooldown limitations. Specifically, Figures 3.4-2 and 3.4-3 are modified to show applicability of the curves up to 11.1 equivalent full-power years (EFPY), to provide revised RT_{NDT} values at 11.1 EFPY at 1/4 and 3/4 thickness, and to indicate the copper content of the controlling material. These changes implement commitments made by North Atlantic Energy Corporation (letter G. Thomas to USNRC dated November 30, 1988) in response to Generic Letter 88-11.

Bases Section 3/4 4.9 is revised to reflect the methodology applied to determine the P/T limits.

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-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 19 to NPF-86
2. Safety Evaluation

cc w/enclosures:
See next page *See previous concurrence

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DATE	4/7/93	4/7/93	4/7/93	3/30/93	/ /

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

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Albert W. De Agazio

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cc w/enclosures:
See next page

cc:

Thomas Dignan, Esq.
A. Ritsher, Esq.
Ropes and Gray
One International Place
Boston, Massachusetts 02110-2624

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

Resident Inspector
U.S. Nuclear Regulatory Commission
Seabrook Nuclear Power Station
Post Office Box 1149
Seabrook, New Hampshire 03874

North Atlantic Energy Service
Corporation
Seabrook Station
Route 1, Lafayette Road
Post Office Box 300
Seabrook, New Hampshire 03874
ATTN: Ted C. Feigenbaum

Mr. T. L. Harpster
North Atlantic Energy Service
Corporation
Post Office Box 300
Seabrook, New Hampshire 03874

Town of Exeter
10 Front Street
Exeter, New Hampshire 03823

Gerald Garfield, Esq.
Day, Berry and Howard
City Place
Hartford, Connecticut 06103-3499

Mr. R. M. Kacich
Northeast Utilities Service Company
Post Office Box 270
Hartford, Connecticut 06141-0270

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

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

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

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

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

Mr. Jack Dolan
Federal Emergency Management Agency
Region I
J.W. McCormack Post Office &
Courthouse Building, Room 442
Boston, Massachusetts 02109

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

John P. Arnold, Attorney General
G. Dana Bisbee, Associate Attorney
General
Attorney General's Office
25 Capitol Street
Concord, New Hampshire 03301

Congressman Nicholas Mavroules
U.S. House of Representatives
70 Washington Street
Salem, Massachusetts 01970

Mr. Robert Sweeney
Bethesda Licensing Office
Suite 610
3 Metro Center
Bethesda, Maryland 20814

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.19
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 August 17, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

   D. The issuance of this 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 (NAESCO) is authorized to act as agent for the: North Atlantic Energy Corporation, Canal Electric Company, The Connecticut Light and Power Company, EUA 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, 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. 19 , 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 the date of its issuance, to be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Albert W. De Agazio for

John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 7, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 19

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
*ix	*ix
x	x
3/4 4-31	3/4 4-31
3/4 4-32	3/4 4-32
B 3/4 4-7	B 3/4 4-7
B 3/4 4-8	B 3/4 4-8
*B 3/4 4-9	*B 3/4 4-9
B 3/4 4-10	B 3/4 4-10

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

Correction letter of 9-12-91

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3.0/4.0 BASES

Controlling material: Base metal
Copper content: 0.06 WT%
RT_{NDT} initial: 40°F
RT_{NDT} after 11.1 EFPY: 1/4T, 108°F
3/4T, 86°F

Curve applicable for heatup rates up to 60°F/hr for the service period up to 11.1 EFPY and contains margins of 10°F and 60 psig for possible instrument errors

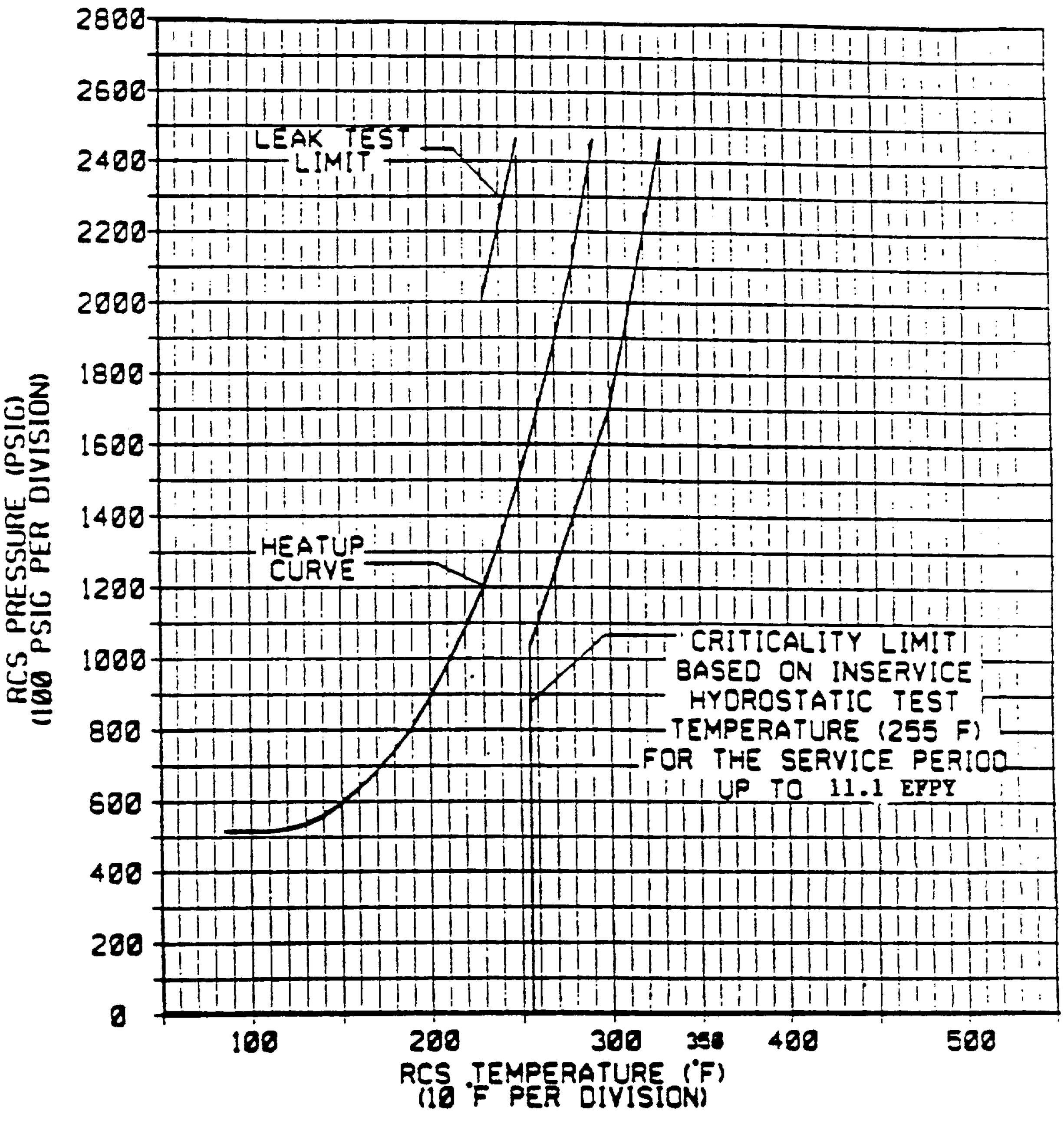

FIGURE 3.4-2

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - APPLICABLE UP TO 11.1 EFPY

MATERIAL PROPERTY BASI

Controlling material:	Base metal
Copper content:	0.06 WT%
RT_{NDT} initial:	40°F
RT_{NDT} after 11.1 EFPY:	1/4T, 108°F
	3/4T, 86°F

Curve applicable for cooldown rates up to 100°F/hr for the service period up to 11.1 EFPY and contains margins of 10°F and 60 psig for possible instrument errors

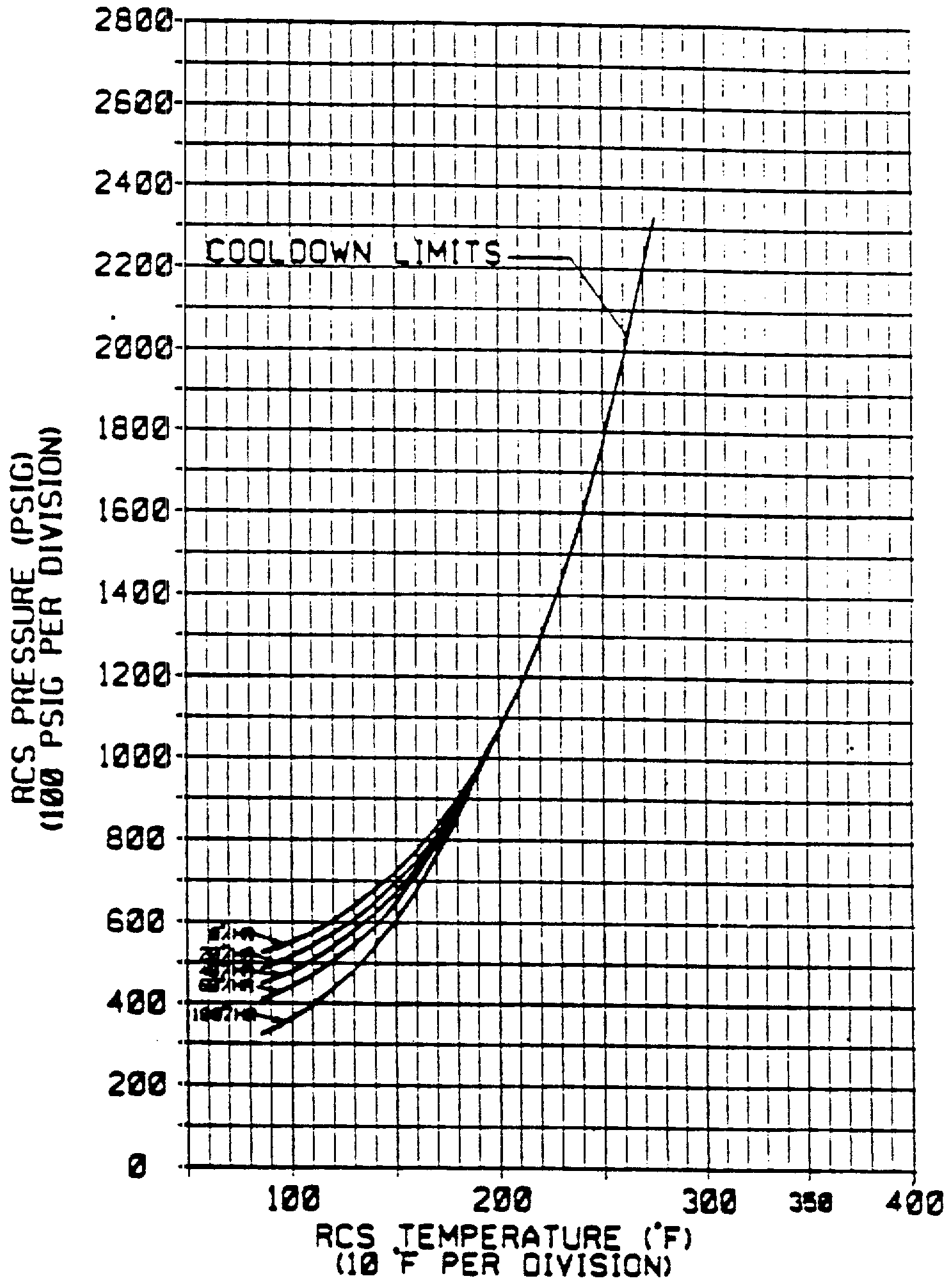

FIGURE 3.4-3

REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - APPLICABLE UP TO 11.1 EFPY

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G:

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 for the service period specified thereon:

    a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

    b. Figures 3.4-2 and 3.4-3 define limits to assure prevention ofnon-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/h and 200°F/h, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F, and

5. System preservice hydrotests and inservice leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Regulatory Guide 1.99, Revision 2, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1972 Winter Addenda to Section III of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves," April 1975.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT} at the end of 11.1 effective full power years (EFPY) of service life. The 11.1 EFPY service life period

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, copper content, and nickel content of the material in question, can be predicted using Figure B 3/4.4-1 and the value of ΔRT_{NDT} computed by Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 11.1 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments.

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-73 and 10CFR50, Appendix H. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the reactor vessel. Therefore, the results obtained from the surveillance specimens can be used to predict future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. Evaluation of surveillance capsule data will be conducted in accordance with NRC Regulatory Guide 1.99, Revision 2.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50, and these methods are discussed in detail in WCAP-7924-A.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures, a semielliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T

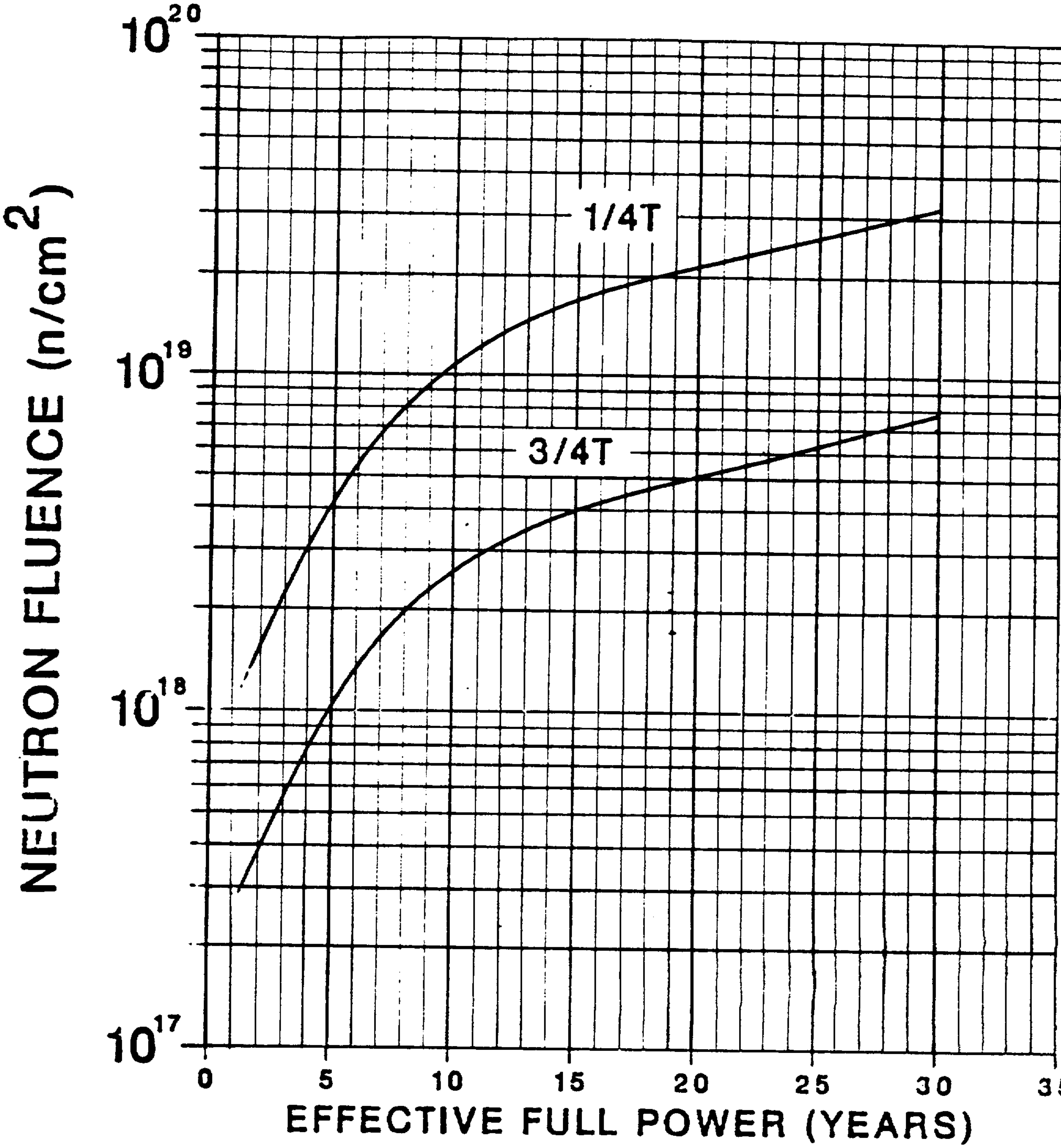

FIGURE B 3/4.4-1

FAST NEUTRON FLUENCE (E>1MeV) AS A FUNCTION OF FULL POWER SERVICE LIFE

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 19 TO FACILITY OPERATING LICENSE NO. NPF-86

NORTH ATLANTIC ENERGY SERVICE CORPORATION

SEABROOK STATION, UNIT NO. 1

DOCKET NO. 50-443

1.0 INTRODUCTION

By letter dated August 17, 1992, the North Atlantic Energy Service Corporation (NAESCO/the licensee) proposed a revision to Section 3/4 4.9 of the Appendix A Technical Specifications (TS) for the Seabrook Station, Unit 1. NAESCO proposed the following: (1) revise the applicability of pressure/temperature (P/T) limits (Figures 3.4-2 and 3.4-3) to state that the curves are valid up to 11.1 effective full power years (EFPY) of operation vice the current 16 EFPY; (2) indicate the copper content of the limiting material, 0.06%; (3) revise the Reference Temperature for Nil Ductility Transition (RT_{ndt}) in the existing P/T limits from 110°F to 108°F and from 87°F to 86°F for the 1/4T (T = reactor vessel beltline thickness) and 3/4T reactor vessel locations, respectively; (4) revise the Bases for TS 3/4 4.9 to state that P/T curves are valid for 11.1 EFPY and that Regulatory Guide (RG) 1.99 (Rev. 2) was used to calculate RT_{ndt}; and (5) delete Bases Figure B 3/4.4-2.

To evaluate the P/T limits, the staff uses the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50, the ASTM Standards and the ASME Code as referenced in Appendix G, 10 CFR 50.36(c)(2), RG 1.99 (Rev. 2), Standard Review Plan (SRP) 5.3.2, and Generic Letter 88-11. Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Effect on Plant Operations," recommends RG 1.99 (Rev. 2) be used in calculating P/T limits, unless the use of different methods can be justified.

Appendices G and H describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in the P/T limits. An acceptable method for constructing the P/T limits is described in SRP 5.3.2. Appendix G specifies fracture toughness and testing requirements for reactor vessel materials under the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested under Appendix H. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature.

Appendix G also requires a prediction of the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART). Generic Letter 88-11 recommended use of the methods in RG 1.99 (Rev. 2), to predict the effect of neutron irradiation on reactor vessel materials. RG 1.99 defines the ART as the sum of the unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

2.0 EVALUATION

The staff has evaluated the effect of neutron irradiation embrittlement on each beltline material in the Seabrook reactor vessel under RG 1.99 (Rev. 2). The staff determined that the limiting material with the highest ART at 11.1 EFPY is the lower shell plate, R1808-3, with 0.06% Cu, 0.57% Ni, and initial RTndt of 40°F. At the 1/4T and 3/4T locations the staff calculated ART values of 107°F and 85°F, respectively, for 11.1 EFPY.

NAESCO used the method in RG 1.99 (Rev. 2), to calculate an ART of 108°F at 1/4T and 86°F at 3/4T for 11.1 EFPY. The difference between licensee's ART and the staff's ART is due to round off error in the staff's calculation of ART values.

Substituting the staff's ARTs into equations in SRP 5.3.2, the staff verified that the proposed P/T limits meet the beltline material requirements in Appendix G.

In addition to beltline materials, Appendix G also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.A.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on the reference temperature of 10°F for the reactor closure flange at the Seabrook Station, Unit 1, the staff determined that the proposed P/T limits satisfies Section IV.A.2 of Appendix G.

NAESCO removed the first surveillance capsules, designated capsule U, during the first refueling outage in August 1991 after 333.37 effective full power days (EFPD) of operation. The results from capsule U were published in Yankee Atomic Electric Company report YAEC-1853. The program utilizes six surveillance capsules. Bases Section 3/4 4/9 has been revised to indicate that NAESCO is committed to removing the remaining surveillance capsules in accordance with the requirements of Appendix H and ASTM E185-73, and to evaluate the data obtained from the surveillance capsules in accordance with RG 1.99 (Rev. 2).

The staff finds that the proposed changes are based on applicable regulatory guidance and conform to the requirements of Appendix G of 10 CFR Part 50. Therefore, the staff finds that the proposed changes are acceptable for incorporation into the Seabrook Unit 1 Technical Specifications.

3.0 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.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. 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 58247). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The 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.

6.0 REFERENCES

1. Regulatory Guide 1.99 (Revision 2), Radiation Embrittlement of Reactor Vessel Materials, May 1988

2. NUREG-0800, Standard Review Plan, Section 5.3.2: Pressure-Temperature Limits

3. Code of Federal Regulations, Title 10, Part 50, Appendix G, Fracture Toughness Requirements

4. Code of Federal Regulations, Title 10, Part 50, Appendix H, Reactor Vessel Material Surveillance Program Requirements

5. Generic Letter 88-11, NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations, July 12, 1988

6. E.C. Biemiller, R.J. Cacciapouti, Analysis of Seabrook Station Unit 1 Reactor Vessel Material and its Impact on Plant Operations, July 12, 1988

7. August 17, 1992, Letter from Ted C. Feigenbaum (NYN-92111) to USNRC Document Control Desk, Subject: License Amendment Request 92-06; Revised RCS Pressure/Temperature Limits

8. August 17, 1992, Letter from Ted C. Feigenbaum (NYN-92112) to USNRC Document Control Desk, Subject: Revised Reference Temperature Values for Pressurized Thermal Shock Events

9. August 17, 1992, Letter from Ted C. Feigenbaum (NYN-92113) to USNRC Document Control Desk, Subject: Reactor Vessel Surveillance Capsule Report

Principal Contributor: A. Wilford

Date: April 7, 1993