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

May 22, 2000

Mr. Ted C. Feigenbaum
Executive Vice President and
Chief Nuclear Officer
North Atlantic Energy Service Corporation
c/o Mr. James M. Peschel
P.O. Box 300
Seabrook, NH 03874

SUBJECT: SEABROOK STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: EDITORIAL AND ADMINISTRATIVE CHANGES TO THE TECHNICAL SPECIFICATIONS (TAC NO. MA4530)

Dear Mr. Feigenbaum:

The Commission has issued the enclosed Amendment No. 70 to Facility Operating License No. NPF-86 for the Seabrook Station, Unit No 1, in response to your application dated December 16, 1998.

The amendment proposes several editorial and administrative changes to the following sections of the Technical Specifications (TSs), Index Page vi, "Figures 3.4-2 and 3.4-3"; Index Page xv, "6.0 Administrative Controls"; 4.2.4.2b, "Determination of Quadrant Power Tilt Ratio"; 6.4.1.7b, "SORC Responsibilities"; 6.4.2.2d, "Station Qualified Reviewer Program"; 6.3.1, "Training"; 6.4.3.9c, "Records of NSARC"; 6.8.1.6.b.1, "Core Operating Limits Report"; and 6.8.1.6.b.10, "Core Operating Limits Report". In addition the following Bases Sections have been revised: Bases 2.2.1, "Reactor Trip System Instrumentation Setpoints"; Bases 3/4.2.4, "Quadrant Power Tilt Ratio"; Bases 3/4.2.5, "DNB Parameters"; Bases 3/4.4.8, "Specific Activity"; and Bases 3/4.5.1, "Accumulators".

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

Sincerely,

Robert N Pulsifer

Robert Pulsifer, Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-443

Enclosures: 1. Amendment No. 70 to NPF-86
2. Safety Evaluation

cc w/encls: See next page

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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. 70
License No. NPF-86

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

   A. The application for amendment filed by the North Atlantic Energy Service Corporation, et al. (the licensee), dated December 16, 1998, 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, Great Bay Power Corporation, Hudson Light & Power Department, Massachusetts Municipal Wholesale Electric Company, Little Bay Power Corporation, New England Power Company, New Hampshire Electric Cooperative, Inc., Taunton Municipal Light Plant, 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) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 70 and the Environmental Protection Plan contained in Appendix B are incorporated into Facility License No. NPF-86. NAESCO shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance, and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

B.C. Buckley for

James W. Clifford, Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: May 22, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 70

FACILITY OPERATING LICENSE NO. NPF-86

DOCKET NO. 50-443

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

Remove	Insert
v*	v*
vi	vi
xv	xv
B 2-7	B 2-7
B 2-8*	B 2-8*
3/4 2-9	3/4 2-9
3/4 2-10*	3/4 2-10*
B 3/4 2-3	B 3/4 2-3
B 3/4 2-4	B 3/4 2-4
B 3/4 4-5*	B 3/4 4-5*
B 3/4 4-6	B 3/4 4-6
B 3/4 5-1	B 3/4 5-1
B 3/4 5-2*	B 3/4 5-2*
6-5	6-5
6-6*	6-6*
6-8	6-8
6-8A	6-8A
6-8B*	6-8B*
6-11	6-11
6-18A	6-18A
6-18B*	6-18B*
6-18C	6-18C

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

6.0 ADMINISTRATIVE CONTROLS

SECTION PAGE

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump Bus trips provide core protection against DNB as a result of complete loss of forced coolant flow. The specified Setpoints assure a Reactor trip signal is generated before the Low Flow Trip Setpoint is reached. Time delays are incorporated in the Underfrequency and Undervoltage trips to prevent spurious Reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the Reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 1.5 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the Reactor trip breakers after the Underfrequency Trip Setpoint is reached shall not exceed 0.6 second. On decreasing power the Undervoltage and Underfrequency Reactor Coolant Pump Bus trips are automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, the Undervoltage and Underfrequency Reactor Coolant Pump Bus trips are reinstated automatically by P-7.

Turbine Trip

A Turbine trip initiates a Reactor trip. On decreasing power, the Reactor trip from the Turbine trip is automatically blocked by P-9 (a power level of approximately 20% of RATED THERMAL POWER); and on increasing power, the Reactor trip from the Turbine trip is reinstated automatically by P-9.

Safety Injection Input from ESF

If a Reactor trip has not already been generated by the Reactor Trip System instrumentation, the ESF automatic actuation logic channels will initiate a Reactor trip upon any signal which initiates a Safety Injection. The ESF instrumentation channels that initiate a Safety Injection signal are shown in Table 3.3-3.

Reactor Trip System Interlocks

The Reactor Trip System interlocks perform the following functions:

P-6 On increasing power, P-6 allows the manual block of the Source Range trip (i.e., prevents premature block of Source Range trip). On decreasing power, Source Range Level trips are automatically reactivated and high voltage is restored.

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Reactor Trip System Interlocks (Continued)

P-7 On increasing power, P-7 automatically enables Reactor trips on low flow in more than one reactor coolant loop, reactor coolant pump bus undervoltage and underfrequency, pressurizer low pressure, and pressurizer high level. On decreasing power, the above listed trips are automatically blocked.

P-8 On increasing power, P-8 automatically enables Reactor trips on low flow in one or more reactor coolant loops. On decreasing power, the P-8 automatically blocks the above trip.

P-9 On increasing power, P-9 automatically enables Reactor trip on Turbine trip. On decreasing power, P-9 automatically blocks Reactor trip on Turbine trip.

P-10 On increasing power, P-10 allows the manual block of the Intermediate Range trip and the Low Setpoint Power Range trip; and automatically blocks the Source Range trip and deenergizes the Source Range high voltage power. On decreasing power, the Intermediate Range trip and the Low Setpoint Power Range trip are automatically reactivated. Provides input to P-7.

P-13 Provides input to P-7.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. The design limit DNBR includes margin to offset any rod bow penalty. Margin is also maintained between the safety analysis limit DNBR and the design limit DNBR. This margin is available for plant design flexibility.

When an F_Q measurement is taken, an allowance for both measurement error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full-core map taken with the movable incore detectors, while 5.21% is appropriate for surveillance results determined with the fixed incore detectors. A 3% allowance is appropriate for manufacturing tolerance.

For operation with the Fixed Incore Detector System (FIDS) Alarm OPERABLE, the cycle-dependent normalized axial peaking factor, K(Z), specified in COLR accounts for axial power shape sensitivity in the LOCA analysis. Assurance that the $F_Q(Z)$ limit on Specification 3.2.2 is met during normal operation and in the event of xenon redistribution following power changes is provided by the FIDS Alarm through the plant process computer. This assures that the consequences of a LOCA would be within specified acceptance criteria.

For operation with the FIDS Alarm inoperable, the cycle-dependent normalized axial peaking factor, K(Z), specified in COLR accounts for possible xenon redistribution following power changes in addition to axial power shape sensitivity in the LOCA analysis. This assures that the consequences of a LOCA would be within specified acceptance criteria.

When RCS $F^N_{\Delta H}$ is measured, no additional allowances are necessary prior to comparison with the established limit. A bounding measurement error of 4.13% for $F^N_{\Delta H}$ has been allowed for in determination of the design DNBR value.

3/4.2.4 QUADRANT POWER TILT RATIO

The purpose of this specification is to detect gross changes in core power distribution between monthly Incore Detector System surveillances. During normal operation the QUADRANT POWER TILT RATIO is set equal to 1.0 once acceptability of core peaking factors has been established by review of incore surveillances. The limit of 1.02 is established as an indication that the power distribution has changed enough to warrant further investigation.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters is maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the updated FSAR assumptions and have been analytically demonstrated adequate to assure compliance with acceptance criteria for each analyzed transient. Operating procedures include allowances for measurement and indication uncertainty so that the limits of 594.3°F for T_{avg} and 2185 psig for pressurizer pressure are not exceeded.

RCS flow must be greater than or equal to, 1) the Thermal Design Flow (TDF) with an allowance for measurement uncertainty and, 2) the minimum measured flow used in place of the TDF in the analysis of the DNB related events when the Revised Thermal Design Procedure (RTDP) methodology is utilized.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

The periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the specified limit.

BASES

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady-State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady-State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady-State Limits.

The Surveillance Requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of 10 CFR Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady-state reactor-to-secondary steam generator leakage rate of 1 gpm. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Seabrook site, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

3/4.4.8 SPECIFIC ACTIVITY (Continued)

The sample analysis for determining the gross specific activity and $\bar{E}$ can exclude the radioiodines because of the low reactor coolant limit of 1 microCurie/gram DOSE EQUIVALENT I-131, and because, if the limit is exceeded, the radioiodine level is to be determined every 4 hours. If the gross specific activity level and radioiodine level in the reactor coolant were at their limits, the radioiodine contribution would be approximately 1%. In a release of reactor coolant with a typical mixture of radioactivity, the actual radioiodine contribution would probably be about 20%. The exclusion of radionuclides with half-lives less than 10 minutes from these determinations has been made for several reasons. The first consideration is the difficulty to identify short-lived radionuclides in a sample that requires a significant time to collect, transport, and analyze. The second consideration is the predictable delay time between the postulated release of radioactivity from the reactor coolant to its release to the environment and transport to the SITE BOUNDARY, which is relatable to at least 30 minutes' decay time. The choice of 10 minutes for the half-life cutoff was made because of the nuclear characteristics of the typical reactor coolant radioactivity. The radionuclides in the typical reactor coolant have half-lives of less than 4 minutes or half-lives of greater than 14 minutes, which allows a distinction between the radionuclides above and below a half-life of 10 minutes. For these reasons the radionuclides that are excluded from consideration are expected to decay to very low levels before they could be transported from the reactor coolant to the SITE BOUNDARY under any accident condition.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture, since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System (RCS) accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration, and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met.

In MODES 1 and 2, the accumulator power-operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In MODES 1, 2, 3, and in MODE 4 within 12 hours of entry into MODE 3 from 4, the accumulator isolation valves are open with their power removed whenever pressurizer pressure is greater than 1000 psig. In addition, as these accumulator isolation valves fail to meet single-failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single-failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double-ended break of the largest RCS cold-leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps and safety injection pumps except the required OPERABLE charging pump to be inoperable in MODES 4 and 5 and in MODE 6 with the reactor vessel head on provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV or RHR suction relief valve.

3/4.5.2 and 3/4.5.3 [illegible]

When the RCS has a vent area equal to or greater than 18 square inches, one Safety Injection pump may be made OPERABLE while in MODE 5 or MODE 6 (below 200°F). When operating in this configuration, cold overpressure protection is provided by the mechanical vent opening, equal to or greater than 18 square inches, that is required to be present in the RCS boundary prior to making the SI pump OPERABLE. This required RCS vent area and the surveillance requirement to verify the presence of the RCS vent area provides assurance that a mass addition transient can be relieved and that adequate cold overpressure protection is provided.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. With the exception of the operating centrifugal charging pump, the ECCS pumps are normally in a standby, non-operating mode. As such, flow path piping has the potential to develop voids and pockets of entrained gases. Maintaining the piping from the refueling water storage tank (RWST) to the RCS full of water (by verifying at the accessible ECCS piping high points and pump casings, excluding the operating centrifugal charging pump) ensures that the system will perform properly, injecting its full capacity into the RCS upon demand. This will also prevent water hammer, pump cavitation, and pumping of non-condensable gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following a safety injection (SI) signal or during shutdown cooling. The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the ECCS piping and the procedural controls governing system operation. Surveillance Requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

3/4.5.4 REFUELING WATER STORAGE TANK

The OPERABILITY of the refueling water storage tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core and (2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

6.2.3 INDEPENDENT TECHNICAL REVIEWS

A Technical Review Program shall be established, implemented and maintained to encompass the following Technical Review responsibilities.

FUNCTION

6.2.3.1 The Technical Review Program responsibilities shall encompass:

a. NRC issuances, industry advisories, Licensee Event Reports, and other sources that may indicate areas for improving plant safety;

b. Internal and external operating experience information that may indicate areas for improving plant safety;

c. Plant operating characteristics, plant operations, modifications, maintenance and surveillance to verify independently that these activities are performed safely and correctly and that human errors are reduced as much as practical, and

d. Making detailed recommendations to the Senior Site Official for procedure revisions, equipment modifications or other means of improving nuclear safety and plant reliability.

The Technical Review Program shall utilize several on-site personnel who are independent of the plant management chain to perform the reviews.

RECORDS

6.2.3.2 Written records of technical reviews shall be maintained. As a minimum these records shall include the results of the activities conducted, the status of recommendations made pursuant to Specification 6.2.3.1 and an assessment of company operations related to the reviews performed.. A copy of the monthly Technical Review Program report shall be provided to the Senior Site Official.

QUALIFICATIONS

6.2.3.3 Personnel performing reviews pursuant to Technical Specification 6.2.3.1 shall have either a bachelor's degree in engineering or related science and at least 2 years professional level experience, at least 1 year of which shall be in the nuclear field, or equivalent education and experience as defined in ANSI/ANS 3.1, 1981, Section 4.1.

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall provide advisory technical support to the Control Room Commander in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the station.

6.3 TRAINING

6.3.1 (THIS SPECIFICATION NUMBER IS NOT USED)

6.4 REVIEW AND AUDIT

6.4.1 STATION OPERATION REVIEW COMMITTEE (SORC)

FUNCTION

6.4.1.1 The SORC shall function to advise the Station Director on all matters related to nuclear safety.

COMPOSITION

6.4.1.2 The SORC shall, as a minimum, be composed of the Chairman and nine individuals who collectively have experience and expertise in the following areas:

Nuclear Power Plant Administrative Controls
Mechanical Maintenance
Electrical Maintenance
Instrumentation & Control
Chemistry
Health Physics
Operations
Technical Support/Engineering
Reactor Engineering

The Station Director shall serve as Chairman of the SORC and shall appoint the SORC members in writing. Members shall have a minimum of eight years power plant experience of which a minimum of three years shall be nuclear power experience. At least one member shall have an SRO license for Seabrook Station.

ALTERNATES

6.4.1.3 All alternate members shall be appointed in writing by the SORC Chairman to serve on a temporary basis and shall have qualifications equivalent to those of the members.

MEETING FREQUENCY

6.4.1.4 The SORC shall meet at least once per calendar month and as convened by the SORC Chairman or one of his designated alternate(s).

QUORUM

6.4.1.5 The quorum of the SORC necessary for the performance of the SORC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or one of his designated alternate(s) and sufficient SORC members including alternates to equal at least 50 percent of the SORC composition.

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1, above 50% of RATED THERMAL POWER*.

ACTION:

With the QUADRANT POWER TILT RATIO determined to exceed 1.02:

a. Within 2 hours reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

b. Within 24 hours and every 7 days thereafter, verify that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their limits by performing Surveillance Requirements 4.2.2.2 and 4.2.3.2. THERMAL POWER and setpoint reductions shall then be in accordance with the ACTION statements of Specifications 3.2.2 and 3.2.3.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

a. Calculating the ratio at least once per 7 days when the alarm is OPERABLE, and

b. Calculating the ratio at least once per 12 hours during steady-state operation when the alarm is inoperable.

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range channel inoperable by using the Incore Detector System to confirm indicated QUADRANT POWER TILT RATIO at least once per 12 hours by either:

a. Using the four pairs of symmetric detector locations or

b. Using the Incore Detector System to monitor the QUADRANT POWER TILT RATIO subject to the requirements of Technical Requirement TR20-3.3.3.2.

*See Special Test Exceptions Specification 3.10.2

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the following limits:

a. Reactor Coolant System T_{avg}, ≤ 594.3°F

b. Pressurizer Pressure, ≥ 2185 psig*

c. Reactor Coolant System Flow shall be:

1. ≥ 382,800 gpm**; and,

2. ≥ 392,800 gpm***

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters shown above shall be verified to be within its limits at least once per 12 hours.

4.2.5.2 The RCS flow rate indicators shall be subjected to CHANNEL CALIBRATION at least once per 18 months.

4.2.5.3 The RCS total flow rate shall be determined by a precision heat balance measurement to be within its limit prior to operation above 95% of RATED THERMAL POWER after each fuel loading. The provisions of Specification 4.0.4 are not applicable for entry into MODE 1.

*Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RATED THERMAL POWER.

**Thermal Design Flow. An allowance for measurement uncertainty shall be made when comparing measured flow to Thermal Design Flow.

***Minimum measured flow used in the Revised Thermal Design Procedure.

6.4.1.7 The SORC shall:

a. Recommend in writing to the Station Director approval or disapproval of items considered under Specification 6.4.1.6a. through d;

b. Render determinations in writing with regard to whether or not each item considered under Specification 6.4.1.6a., b. and d. constitutes an unreviewed safety question; and

c. Provide written notification within 24 hours to the Executive Vice President & Chief Nuclear Officer and the NSARC of disagreement between the SORC and the Station Director however, the Station Director shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1.

RECORDS

6.4.1.8 The SORC shall maintain written minutes of each SORC meeting that, at a minimum, document the results of all SORC activities performed under the responsibility provisions of these Technical Specifications. Copies shall be provided to the Executive Vice President & Chief Nuclear Officer and the NSARC.

6.4.2 STATION QUALIFIED REVIEWER PROGRAM

FUNCTION

6.4.2.1 The Station Director may establish a Station Qualified Reviewer Program whereby required reviews of designated procedures or classes of procedures required by Specification 6.4.1.6.a are performed by Station Qualified Reviewers and approved by the designated department heads. These reviews are in lieu of reviews by the SORC. However, procedures which require a 10 CFR 50.59 evaluation must be reviewed by the SORC.

RESPONSIBILITIES

6.4.2.2 The Station Qualified Reviewer Program shall:

a. Provide for the review of designated procedures, programs, and changes thereto by a Qualified Reviewer(s) other than the individual who prepared the procedure, program, or change.

b. Provide for cross-disciplinary review of procedures, programs, and changes thereto when organizations other than the preparing organization are affected by the procedure, program, or change.

c. Ensure cross-disciplinary reviews are performed by a Qualified Reviewer(s) in affected disciplines, or by other persons designated by cognizant department heads as having specific expertise required to assess a particular procedure, program or change. Cross-disciplinary reviewers may function as a committee.

d. Provide for a screening of designated procedures, programs and changes thereto to determine if an evaluation should be performed in accordance with the provisions of 10 CFR 50.59 to verify that an unreviewed safety question does not exist. This screening will be performed by personnel trained and qualified in performing 10 CFR 50.59 screenings.

e. Provide for written recommendation by the Qualified Reviewer(s) to the responsible department head for approval or disapproval of procedures and programs considered under Specification 6.4.1.6a and that the procedure or program was screened by a qualified individual and found not to require a 10 CFR 50.59 evaluation.

6.4.2.3 If the responsible department head determines that a new program, procedure, or change thereto requires a 10 CFR 50.59 evaluation, that designated department head will ensure the required evaluation is performed to determine if the new procedure, program, or change involves an unreviewed safety question. The new procedure, program, or change will then be forwarded with the 10 CFR 50.59 evaluation to SORC for review.

6.4.2.4 Personnel recommended to be Station Qualified Reviewers shall be designated in writing by the Station Director for each procedure, program, or class of procedure or program within the scope of the Station Qualified Reviewer Program.

6.4.2.5 Temporary procedure changes shall be made in accordance with Specification 6.7.3 with the exception that changes to procedures for which reviews are assigned to Qualified Reviewers will be reviewed and approved as described in Specification 6.4.2.2.

RECORDS

6.4.2.6 The review of procedures and programs performed under the Station Qualified Reviewer Program shall be documented in accordance with administrative procedures.

TRAINING AND QUALIFICATION

6.4.2.7 The training and qualification requirements of personnel designated as a Qualified Reviewer in accordance with the Station Qualified Reviewer Program shall be in accordance with administrative procedures. Qualified reviewers shall have:

a. A Bachelors degree in engineering, related science, or technical discipline, and two years of nuclear power plant experience;

OR

b. Six years of nuclear power plant experience;

OR

c. An equivalent combination of education and experience as approved by the designated department head.

6.4.3 NUCLEAR SAFETY AUDIT REVIEW COMMITTEE (NSARC)

FUNCTION

6.4.3.1 The NSARC shall function to provide independent review and audit of designated activities. The NSARC shall report to and advise the Executive Vice President & Chief Nuclear Officer on those areas of responsibility specified in Specifications 6.4.3.7 and 6.4.3.8.

COMPOSITION

6.4.3.2 The NSARC shall be composed of at least five (5) individuals. The Chairman, Vice Chairman and members, including designated alternates, shall be appointed in writing by the Executive Vice President & Chief Nuclear Officer. Collectively, the individuals appointed to the NSARC should have experience and expertise in the following areas:

a. Nuclear power plant operations,
b. Nuclear engineering,
c. Chemistry and radiochemistry,
d. Metallurgy,
e. Instrumentation and control,
f. Radiological safety,
g. Mechanical and electrical engineering, and
h. Quality assurance practices.

Each member shall meet the qualifications of ANSI 3.1-1978, Section 4.7.

ALTERNATES

6.4.3.3 All alternate members shall be appointed in writing by the Executive Vice President & Chief Nuclear Officer to serve on a temporary basis; however, no more than a minority shall participate as voting members in NSARC activities at any one time.

CONSULTANTS

6.4.3.4 Consultants shall be utilized as determined by the NSARC to provide expert advice to the NSARC.

MEETING FREQUENCY

6.4.3.5 The NSARC shall meet at least once per 6 months ± 6 weeks.

QUORUM

6.4.3.6 The quorum of the NSARC necessary for the performance of the NSARC review and audit functions of these Technical Specifications shall consist of the Chairman or Vice-Chairman and at least four NSARC members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the unit. The Vice Chairman, or his designated alternate, can participate as an NSARC member when the Chairman is in attendance.

RECORDS

6.4.3.9 Records of NSARC activities shall be prepared and distributed as indicated below:

a. Minutes of each NSARC meeting shall be prepared and forwarded to Executive Vice President & Chief Nuclear Officer within 30 working days following each meeting;

b. Reports of reviews encompassed by Specification 6.4.3.7 shall be included in the minutes where applicable or forwarded under separate cover to the Executive Vice President & Chief Nuclear Officer within 30 working days following completion of the review; and

c. Audit reports encompassed by Specification 6.4.3.8 shall be forwarded to the Executive Vice President & Chief Nuclear Officer and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

6.5 REPORTABLE EVENT ACTION

The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the SORC and the results of this review shall be submitted to the NSARC and the Executive Vice President & Chief Nuclear Officer.

6.6 SAFETY LIMIT VIOLATION

The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Executive Vice President & Chief Nuclear Officer and the NSARC shall be notified within 24 hours;

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the SORC. This report shall describe: (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems, or structures, and (3) corrective action taken to prevent recurrence;

c. The Safety Limit Violation Report shall be submitted to the Commission, the NSARC, and the Executive Vice President & Chief Nuclear Officer within 14 days of the violation; and

d. Operation of the station shall not be resumed until authorized by the Commission.

6.8.1.6.a. (Continued)

5. Shutdown Rod Insertion limit for Specification 3.1.3.5,
6. Control Rod Bank Insertion limits for Specification 3.1.3.6,
7. AXIAL FLUX DIFFERENCE limits for Specification 3.2.1,
8. Heat Flux Hot Channel Factor, F^{RTP}_{Q} and K(Z) for Specification 3.2.2,
9. Nuclear Enthalpy Rise Hot Channel Factor, and $F^{RTP}_{\Delta H}$ for Specification 3.2.3.

The CORE OPERATING LIMITS REPORT shall be maintained available in the Control Room.

6.8.1.6.b The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

1. WCAP-10266-P-A, Rev. 2 with Addenda (Proprietary) and WCAP-11524-A, Rev. 2 with Addenda (Nonproprietary), "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code", March, 1987.

   Methodology for Specification:
   3.2.2 - Heat Flux Hot Channel Factor

2. WCAP-10079-P-A (Proprietary) and WCAP-10080-A (Nonproprietary), "NOTRUMP: A Nodal Transient Small Break and General Network Code", August, 1985

   Methodology for Specification:
   3.2.2 - Heat Flux Hot Channel Factor

3. YAEC-1363-A - "CASMO-3G Validation," April 1988.

   YAEC-1659-A, "SIMULATE-3 Validation and Verification," September 1988.

   Methodology for Specifications:
   3.1.1.1 - SHUTDOWN MARGIN for MODES 1, 2, 3, and 4
   3.1.1.2 - SHUTDOWN MARGIN for MODE 5
   3.1.1.3 - Moderator Temperature Coefficient
   3.1.3.5 - Shutdown Rod Insertion Limit
   3.1.3.6 - Control Rod Insertion Limits
   3.2.1 - AXIAL FLUX DIFFERENCE
   3.2.2 - Heat Flux Hot Channel Factor
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

4. Seabrook Station Updated Final Safety Analysis Report, Section 15.4.6, "Chemical and Volume Control System Malfunction That Results in a Decrease in the Boron Concentration in the Reactor Coolant System".

   Methodology for Specifications:
   3.1.1.1 - SHUTDOWN MARGIN for MODES 1, 2, 3, and 4
   3.1.1.2 - SHUTDOWN MARGIN for MODE 5

6.8.1.6.b. (Continued)

5. YAEC-1241, "Thermal-Hydraulic Analysis of PWR Fuel Elements Using the CHIC-KIN Code", R. E. Helfrich, March 1981

   Methodology for Specification:
   3.2.1 - AXIAL FLUX DIFFERENCE
   3.2.2 - Heat Flux Hot Channel Factor
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

6. YAEC-1849P, "Thermal-Hydraulic Analysis Methodology Using VIPRE-01 For PWR Applications, "October 1992

   Methodology for Specification:
   2.2.1 - Limiting Safety System Settings
   3.2.1 - AXIAL FLUX DIFFERENCE
   3.2.2 - Heat Flux Hot Channel Factor
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

7. YAEC-1854P, "Core Thermal Limit Protection Function Setpoint Methodology For Seabrook Station, "October 1992

   Methodology for Specification:
   2.2.1 - Limiting Safety System Settings
   3.1.3.5 - Shutdown Rod Insertion Limit
   3.1.3.6 - Control Rod Insertion Limits
   3.2.1 - AXIAL FLUX DIFFERENCE
   3.2.2 - Heat Flux Hot Channel Factor
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

8. YAEC-1856P, "System Transient Analysis Methodology Using RETRAN for PWR Applications," December 1992

   Methodology for Specification:
   2.2.1 - Limiting Safety System Settings
   3.1.1.3 - Moderator Temperature Coefficient
   3.1.3.5 - Shutdown Rod Insertion Limit
   3.1.3.6 - Control Rod Insertion Limits
   3.2.1 - AXIAL FLUX DIFFERENCE
   3.2.2 - Heat Flux Hot Channel Factor
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

9. YAEC-1752, "STAR Methodology Application for PWRs, Control Rod Ejection, Main Steam Line Break," October 1990

   Methodology for Specification:
   3.1.1.3 - Moderator Temperature Coefficient
   3.1.3.5 - Shutdown Rod Insertion Limit
   3.1.3.6 - Control Rod Insertion Limits
   3.2.1 - AXIAL FLUX DIFFERENCE
   3.2.2 - Heat Flux Hot Channel Factor
   3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

6.8.1.6.b. (Continued)

10. YAEC-1855PA, "Seabrook Station Unit 1 Fixed Incore Detector System Analysis," October 1992

    Methodology for Specification:
    3.2.1 - AXIAL FLUX DIFFERENCE
    3.2.2 - Heat Flux Hot Channel Factor
    3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

11. YAEC-1624P, "Maine Yankee RPS Setpoint Methodology Using Statistical Combination of Uncertainties - Volume 1 - Prevention of Fuel Centerline Melt," March 1988

    Methodology for Specification:
    3.2.1 - AXIAL FLUX DIFFERENCE
    3.2.2 - Heat Flux Hot Channel Factor
    3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor

12. NYN-95048, Letter from T. C. Feigenbaum (NAESCo) to NRC, "License Amendment Request 95-05: Positive Moderator Temperature Coefficient", May 30, 1995

    Methodology for Specification:
    3.1.1.3- Moderator Temperature Coefficient

13. WCAP-12610-P-A, "VANTAGE + Fuel Assembly Reference Core Report", April 1995, (Westinghouse Proprietary)

    Methodology for Specification:
    3.2.2 - Heat Flux Hot Channel Factor

6.8.1.6.c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as SHUTDOWN MARGIN, and transient and accident analysis limits) of the safety analysis are met. The CORE OPERATING LIMITS REPORT for each reload cycle, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, to the NRC Document Control Desk with copies to the Regional Administrator and the Resident Inspector.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 70 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 December 16, 1998, the North Atlantic Energy Service Corporation (North Atlantic) submitted License Amendment Request (LAR) 98-09 which requested changes to the Seabrook Station Technical Specifications (TSs). The amendment proposes several editorial and administrative changes to the following sections of the TSs, Index Page vi, "Figures 3.4-2 and 3.4-3"; Index Page xv, "6.0 Administrative Controls"; 4.2.4.2b, "Determination of Quadrant Power Tilt Ratio"; 6.4.1.7b, "SORC Responsibilities"; 6.4.2.2d, "Station Qualified Reviewer Program"; 6.3.1, "Training"; 6.4.3.9c, "Records of NSARC"; 6.8.1.6.b.1, "Core Operating Limits Report"; and 6.8.1.6.b.10, "Core Operating Limits Report". In addition the following Bases Sections have been revised: Bases 2.2.1, "Reactor Trip System Instrumentation Setpoints"; Bases 3/4.2.4, "Quadrant Power Tilt Ratio"; Bases 3/4.2.5, "DNB Parameters"; Bases 3/4.4.8, "Specific Activity"; and Bases 3/4.5.1, "Accumulators".

2.0 BACKGROUND

The licensee has proposed five changes to the administrative section of the TS, four editorial changes, and four changes to the Bases Sections. The licensee stated that the proposed changes revise references and statements that are inaccurate or provide relief from administrative controls which provide insignificant safety benefit.

3.0 EVALUATION

3.1 TS 6.4.1.7b Station Operation Review Committee (SORC) Responsibilities

This administrative change is to exclude reference to TS 6.4.1.6c. and e. from TS 6.4.1.7b. Presently, TS 6.4.1.7b requires the SORC to render determinations in writing with regard to whether or not each item considered under Specification 6.4.1.6a. through e. constitutes an unreviewed safety question (USQ).

TS 6.4.1.6c. specifies that the SORC shall be responsible for review of all proposed changes to Appendix “A” TSs. TS 6.4.1.7b. requires the SORC to render a determination as to whether proposed changes to Appendix “A” TSs constitutes a USQ. All proposed changes to

Appendix "A" TSs are made in accordance with 10 CFR 50.90, 50.91, and 50.92, which require NRC staff approval prior to implementation. The NRC staff amends licenses based on its determination that no significant hazards are associated with the proposed change and based on its safety evaluation of the proposed change. The requirement under TS 6.4.1.7b. for the SORC to render determinations as to whether proposed changes to Appendix "A" TSs constitute a USQ is an administrative control that provides insignificant safety benefit. Regardless if a USQ exists or not, all changes to TSs must be reviewed by the SORC and receive NRC staff review and approval prior to implementation. The staff agrees that the appropriate reviews and approvals are in place for TS changes whether or not the SORC renders a determination in writing as to whether or not proposed changes to Appendix "A" TSs constitute a USQ. The staff agrees that this proposed TS change to remove the reference to TS 6.4.1.6.c is administrative in nature and, therefore, is acceptable.

TS 6.4.1.6e. requires the SORC to investigate all violations of the TSs. After the investigation, SORC prepares and forwards a report to the Executive Vice President & Chief Nuclear Officer and to the Nuclear Safety Audit Review Committee (NSARC), providing an evaluation, and recommendations regarding the violation to prevent recurrence. In addition, TS 6.4.1.6e, requires the SORC to determine, in writing, whether the violation of the TSs constitutes a USQ. Safe operation of the facility is governed by the Limiting Conditions For Operation (LCOs) and associated Actions and SRs stated in the TSs, as required by the provisions of 10 CFR 50.36. When an LCO is not met, 10 CFR 50.36 requires the licensee to shut down the reactor or follow any remedial action permitted by the TSs until the condition can be met. While a potential USQ condition exist when the plant is operating at the time of the TS violation, compliance with the requirements of the TSs ensures continued safe operation of the facility. In addition, any operation or condition prohibited by the TSs require reporting pursuant to the provisions of 10 CFR 50.73. Therefore, the determination whether a USQ existed at the time of the TS violation is, essentially, a retrospective review of a temporary condition which provides little benefit to the continued safe operation of the facility.

Should a licensee desire to operate the facility outside the requirements currently specified in the TSs, then the licensee must either request an amendment to the current Operating License, pursuant to the provisions of 10 CFR 50.90, 50.91, and 50.92, or seek issuance of a Notice of Enforcement Discretion (NOED), pursuant to NUREG-1600, "General Statement of Policy and Procedures for NRC Enforcement Actions," (with a supporting safety basis for the temporary condition and, if permanent, a follow up license amendment request), both of which require NRC staff review and approval for continued operation.

The SORC's responsibility to investigate TS violations and make appropriate recommendations to prevent any recurrence are maintained. The requirement to review TS violations that occur temporarily to determine if a USQ existed at the time of the TS violation directs the SORC away from its primary responsibilities and provides little benefit to ensure continued safe operation of the facility. The staff agrees that the removal of the reference to TS 6.4.1.6e is administrative in nature and, therefore, is acceptable.

3.2 <u>TS 6.4.2.2d Station Qualified Reviewer Program</u>

This administrative change is to TS 6.4.2.2d. to reflect that Station Qualified Reviewers need only be trained and qualified to perform 10 CFR 50.59 screenings in lieu of 10 CFR 50.59 evaluations. The proposed change would make TS 6.4.2.2d. consistent with TS 6.4.2.2e. which states ". . . that the procedure or program was screened by a qualified individual and found not to require a 10 CFR 50.59 evaluation." The Station Qualified Reviewer Program, as outlined in TS 6.4.2, has provision for qualified individuals, who perform reviews of designated procedures, programs, and changes considered under TS 6.4.1.6a, to perform 10 CFR 50.59 screenings. A screening is made to determine if further evaluation, under the provisions of 10 CFR 50.59, is warranted in the determination that an unreviewed safety question does or does not exist. To perform 10 CFR 50.59 screenings, it is not necessary that these individuals be specifically trained and qualified in performing 10 CFR 50.59 safety evaluations. The licensee acknowledged that these individuals, who perform screenings, must be familiar with the entire 10 CFR 50.59 process. In addition, other individuals specifically trained and qualified to perform 10 CFR 50.59 evaluations may be designated to perform this task. The requirement of ensuring that a 10 CFR 50.59 evaluation is performed is specified in TS 6.4.2.2e. TS 6.4.2.2e. requires that the Qualified Reviewer(s) make a written recommendation to the responsible department head for the approval or disapproval of procedures and programs that are considered under TS 6.4.1.6a. TS 6.4.2.3 requires that the department head make the final determination on whether or not a 10 CFR 50.59 evaluation is necessary. Should the department head determine that a 10 CFR 50.59 review is needed it is his responsibility to ensure that a 10 CFR 50.59 evaluation is performed and presented, along with the new procedure, program, or proposed change, to the SORC for review. TS 6.4.2.1 requires that procedures which require a 10 CFR 50.59 evaluation must be reviewed by the SORC. The level of administrative control in the review of designated procedures, programs, and changes thereto considered under TS 6.4.1.6a remains the same but the Station Qualified Reviewers will need only be trained and qualified to perform 10 CFR 50.59 screenings. The staff concludes that the proposed change provides sufficient controls to ensure that the requirements of 10 CFR Part 50, Appendix B, Criterion VI, "Document Control", are satisfied. Therefore, the change is acceptable.

3.3 TS 6.3.1 Training

This administrative change deletes TS 6.3.1, "Training". TS 6.3.1 requires maintenance of a retraining and replacement licensed training program for the station staff. The reason for removing this requirement is that it does not satisfy any of the four criteria for inclusion in the TSs given in 10 CFR 50.36(c)(2)(ii). In addition, the training/retraining program information contained in TS 6.3.1 is addressed in Updated Final Safety Analysis Report (UFSAR) Chapter 13. Training and re-qualification of those positions are as specified in 10 CFR Part 55 and 10 CFR 50.120, and as delineated in the Institute of Nuclear Power Operations (INPO) Accredited Program Descriptions for licensed training programs. The INPO Program Descriptions are consistent with the 1989 licensing commitment (Letter NYN-89144 from Ted C. Feigenbaum to NRC, "FSAR Section 13.2," dated November 13, 1989) to comply with Regulatory Guide 1.8, Revision 2, as discussed with the NRC staff. Likewise, the UFSAR contains the standards to which station staff personnel are qualified. Changes to the UFSAR are pursuant to the provisions of 10 CFR 50.59, therefore, the level of administrative control for the licensee training program remains comparable. Plant safety will not be adversely affected as a result of deleting TS 6.3.1. The deletion of this TS is consistent with the Improved Standard TSs. Since this requirement does not satisfy any of the four criteria in 10 CFR 50.36(c)(2)(ii) and will be maintained by the UFSAR and their commitment to Regulatory Guide 1.8, the staff concludes that the control of these provisions by 10 CFR 50.59 is sufficient and, therefore, removing TS 6.3.1 is acceptable.

3.4 Index Page vi Figures 3.4-2 and 3.4-3

This editorial change corrects TSs Index Page vi to indicate that the service period, as shown on TS Figures 3.4-2 and 3.4-3 for the Reactor Coolant System (RCS) Heatup/Cooldown Limitation curves, is applicable up to 11.1 effective full power years (EFPY). This change was inadvertently omitted from the submittal of LAR 92-06, "Revised RCS Pressure/Temperature Limits", dated August 17, 1992, when the service period for RCS heatup and cooldown rate curves was revised from 16 EFPY to 11.1 EFPY. The staff approved this change to the service period by License Amendment 19, issued April 7, 1993. The staff agrees that this change is editorial in nature and, therefore, the proposed TS change is acceptable.

3.5 Index Page xv 6.0 Administrative Controls

This editorial change is to TS Index Page xv. This change will add a reference to TS 6.15, Containment Leakage Rate Testing Program in the index. TS 6.15 was previously approved in License Amendment 49, issued February 24, 1997. The reference was inadvertently omitted from the submittal of LAR 96-05, "Implementation of 10 CFR 50 Appendix J, Option B, Containment Leakage Rate Testing (TAC M95312)," dated June 4, 1996. Inclusion of TS 6.15 in the TSs Index is for completeness and will allow the user an easy method of locating TS 6.15 within the TSs Manual. The staff agrees that the proposed change is an editorial change for reference purposes only and, therefore, the proposed TS change is acceptable.

3.6 TS 4.2.4.2b Determination of Quadrant Power Tilt Ratio

This editorial change is to Quadrant Power Tilt Ratio Surveillance Requirement (SR) 4.2.4.2b to change the current reference from TS 3.3.3.2 to Technical Requirement TR20-3.3.3.2. LAR 96-02, "Relocation of Selected TSs to Licensee Controlled Documents - Incore Detector System, Seismic Instrumentation, Meteorological Instrumentation and Turbine Overspeed Protection (TAC M96723)", dated October 17, 1996, and subsequently approved as License Amendment 50, issued March 12, 1997, relocated Incore Detector System TS 3.3.3.2 from TSs to the Seabrook Station Technical Requirements (SSTR) Manual. Therefore, TS 3.3.3.2 is no longer in the TSs. This change inadvertently omitted the change in reference in SR 4.2.4.2b to reflect that the TS section had been relocated. The staff agrees that this change is editorial in nature and, therefore, is acceptable.

3.7 TS 6.4.3.9c Records of NSARC

This editorial change is to TS 6.4.3.9c to revise the reference from TS 6.4.2.8 to TS 6.4.3.8. Seabrook Station's TSs do not contain a TS 6.4.2.8. The staff has confirmed that TS 6.4.3.8 is the correct reference concerning audits. The staff agrees that this change is editorial in nature and, therefore, the proposed TS change is acceptable.

3.8 TS 6.8.1.6.b.1 Core Operating Limits Report

This editorial change is to TS 6.8.1.6.b.1. to revise the date of issue of WCAP-11524-A and issued addenda. The correct issue is WCAP-11524-A, Rev. 2 with Addenda, March 1987. Both issues of the report describe the large break loss-of-coolant-accident (LOCA) methodology used to prepare the LOCA Safety Analysis Report supplied by Westinghouse via letter 93NA*-G-0037, August 31, 1993. This report was submitted in support for the approval of License Amendment 33, issued November 23, 1994, to the TSs. There has been no change in the methodology actually applied since approval of Amendment 33. The staff agrees that this proposed TS change is editorial in nature (update of a reference previously approved) and, therefore, is acceptable.

3.9 TS 6.8.1.6.b.10 Core Operating Limits Report

This editorial change is to TS 6.8.1.6.b.10. to revise the reference to Yankee Atomic Electric Company document number YAEC-1855P to YAEC-1855PA to make the TS consistent with the Core Operating Limits Report (COLR) supporting documentation. This revision reconciles differences between the Cycle 5 COLR references stated in TS 6.8.1.6.b.10. and the references contained in the Seabrook Station Cycle 5 Core Reload Safety Evaluation, YAEC-1925, dated 10/2/95. The reference in YAEC-1925, "YAEC-1855PA," refers to a re-issuance of the submitted YAEC-1855P topical report to the NRC staff signifying that it was approved by the NRC staff. The difference between the two reports is that YAEC-1855PA was modified to include a copy of the NRC staff's SER and TER for YAEC-1855P, as well as copies of the review questions and responses with no changes to the technical content of the report. Preparation of an approved version of the topical report is in keeping with the NRC staff recommendations. These reports describe the Fixed Incore Detector System Analysis methodology which was submitted in support of Amendment No. 27, issued December 22, 1993, to the TSs. The licensee has stated that there has been no change in the methodology actually applied since approval of Amendment 27. The staff agrees that this proposed TS change is editorial in nature and, therefore, is acceptable.

3.10 Bases Section Changes

The following Bases sections have been revised: 2.2.1, "Reactor Trip System Instrumentation Setpoints"; 3/4.2.4, "Quadrant Power Tilt Ratio"; 3/4.2.5, "DNB Parameters"; 3/4.4.8, "Specific Activity"; and 3/4.5.1, "Accumulators". The proposed changes correct errors and provide clarifications to the above Bases sections. All changes are reviewed by a Bases control program which is described in the UFSAR. The staff has no objections to the proposed changes.

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

This amendment changes recordkeeping, reporting, or administrative procedures or requirements. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). 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.

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: A. Wang

Date: May 22, 2000

May 22, 2000

Mr. Ted C. Feigenbaum
Executive Vice President and
Chief Nuclear Officer
North Atlantic Energy Service Corporation
c/o Mr. James M. Peschel
P.O. Box 300
Seabrook, NH 03874

SUBJECT: SEABROOK STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: EDITORIAL AND ADMINISTRATIVE CHANGES TO THE TECHNICAL SPECIFICATIONS (TAC NO. MA4530)

Dear Mr. Feigenbaum:

The Commission has issued the enclosed Amendment No. 70 to Facility Operating License No. NPF-86 for the Seabrook Station, Unit No 1, in response to your application dated December 16, 1998.

The amendment proposes several editorial and administrative changes to the following sections of the Technical Specifications (TSs), Index Page vi, "Figures 3.4-2 and 3.4-3"; Index Page xv, "6.0 Administrative Controls"; 4.2.4.2b, "Determination of Quadrant Power Tilt Ratio"; 6.4.1.7b, "SORC Responsibilities"; 6.4.2.2d, "Station Qualified Reviewer Program"; 6.3.1, "Training"; 6.4.3.9c, "Records of NSARC"; 6.8.1.6.b.1, "Core Operating Limits Report"; and 6.8.1.6.b.10, "Core Operating Limits Report". In addition the following Bases Sections have been revised: Bases 2.2.1, "Reactor Trip System Instrumentation Setpoints"; Bases 3/4.2.4, "Quadrant Power Tilt Ratio"; Bases 3/4.2.5, "DNB Parameters"; Bases 3/4.4.8, "Specific Activity"; and Bases 3/4.5.1, "Accumulators".

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

Sincerely,

/RA/
Robert Pulsifer, Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-443

Enclosures: 1. Amendment No. 70 to NPF-86
2. Safety Evaluation

cc w/encls: See next page

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