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. 49 TO FACILITY OPERATING LICENSE NPF-86: IMPLEMENTATION OF 10 CFR 50 APPENDIX J, OPTION B - LICENSE AMENDMENT REQUEST 96-05 (TAC NO. M95312)

Dear Mr. Feigenbaum:

The Commission has issued the enclosed Amendment No. 49 to Facility Operating License No. NPF-86 for the Seabrook Station, Unit No. 1 (Seabrook), in response to your application (LAR 96-05) dated June 4, 1996, as supplemented by letter dated January 8, 1997.

The amendment revises Seabrook Appendix A Technical Specifications (TSs) 1.7, "Containment Integrity", 3/4.6.1, "Primary Containment", and 3/4.6.5, "Containment Enclosure Building", to incorporate the provisions of Option B to 10 CFR Part 50, Appendix J. TS Section 6.15, "Containment Leakage Rate Testing Program", has been added to establish a Containment Leakage Rate Testing Program, as specified in Regulatory Guide 1.163, dated September 1995, to support these changes. In addition, TS 3.6.1.7 and 4.6.1.7.1 have been revised to incorporate an increased leak testing interval and to include reference to the Containment Leakage Rate Testing Program.

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, 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-96-009

Enclosures: 1. Amendment No. 49 to NPF-86
2. Safety Evaluation

cc w/encls: See next page

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February 24, 1997

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Director - Nuclear Licensing Services
P.O. Box 128
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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. 49
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 June 4, 1996, as supplemented January 8, 1997, 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 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) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 49, 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

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 24, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 49

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
1-1*	1-1*
1-2	1-2
3/4 6-1	3/4 6-1
3/4 6-2	3/4 6-2
3/4 6-3	3/4 6-3
3/4 6-4	3/4 6-4
3/4 6-7	3/4 6-7
3/4 6-8	3/4 6-8
3/4 6-11	3/4 6-11
3/4 6-12	3/4 6-12
3/4 6-13	3/4 6-13
3/4 6-14*	3/4 6-14*
3/4 6-25	3/4 6-25
6-23*	6-23*
6-24	6-24
B 3/4 6-1	B 3/4 6-1
B 3/4 6-2	B 3/4 6-2
B 3/4 6-3*	B 3/4 6-3*
B 3/4 6-4	B 3/4 6-4

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Technical Specification which prescribes remedial measures required under designated conditions.

ACTUATION LOGIC TEST

1.2 An ACTUATION LOGIC TEST shall be the application of various simulated input combinations in conjunction with each possible interlock logic state and verification of the required logic output. The ACTUATION LOGIC TEST shall include a continuity check, as a minimum, of output devices.

ANALOG CHANNEL OPERATIONAL TEST

1.3 An ANALOG CHANNEL OPERATIONAL TEST shall be the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY of alarm, interlock and/or trip functions. The ANALOG CHANNEL OPERATIONAL TEST shall include adjustments, as necessary, of the alarm, interlock and/or Trip Setpoints such that the Setpoints are within the required range and accuracy.

AXIAL FLUX DIFFERENCE

1.4 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.

CHANNEL CALIBRATION

1.5 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel such that it responds within the required range and accuracy to known values of input. The CHANNEL CALIBRATION shall encompass the entire channel including the sensors and alarm, interlock and/or trip functions and may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.6 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or

2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions.

b. All equipment hatches are closed and sealed,

c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,

d. The containment leakage rates are in accordance with the Containment Leakage Rate Testing Program, and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT

1.10 The CORE OPERATING LIMITS REPORT (COLR) provides core operating limits for the current operating reload cycle. The cycle specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.8.1.6. Plant operation within these operating limits is addressed in individual specifications.

DIGITAL CHANNEL OPERATIONAL TEST

1.11 A DIGITAL CHANNEL OPERATIONAL TEST shall consist of exercising the digital computer hardware using data base manipulation and/or injecting simulated process data to verify OPERABILITY of alarm and/or trip functions. The Digital Channel Operational Test definition is only applicable to the Radiation Monitoring Equipment.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions except for valves that are open under administrative control as permitted by Specification 3.6.3; and

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3.

*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited in accordance with the Containment Leakage Rate Testing Program.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the overall integrated leakage rate exceeding 1.0 La, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated in accordance with the Containment Leakage Rate Testing Program.

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LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE in accordance with the Containment Leakage Rate Testing Program.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one containment air lock door inoperable:

   1. Maintain at least the OPERABLE air lock door closed* and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed,

   2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days,

   3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours, and

   4. The provisions of Specification 3.0.4 are not applicable.

b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*Except during entry to repair an inoperable inner door, for a cumulative time not to exceed 1 hour per year.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. With the leakage rate in accordance with the Containment Leakage Rate Testing Program.

b. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

PRIMARY CONTAINMENT

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment vessel shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the structural integrity of the containment vessel not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.6 The structural integrity of the containment vessel shall be determined by a visual inspection of the exposed accessible interior and exterior surfaces of the vessel in accordance with the Containment Leakage Rate Testing Program. Any abnormal degradation of the containment vessel detected during the above required inspections shall be reported to the Commission in a Special Report pursuant to Specification 6.8.2 within 15 days.

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each 8-inch containment purge supply and exhaust isolation valve shall be OPERABLE and sealed closed except when open for purge system operation for pressure control; for ALARA, respirable, and air quality considerations to facilitate personnel entry; and for surveillance tests that require the valve(s) to be open.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one or more of the 8-inch containment purge supply or exhaust isolation valves open for reasons other than given in Specification 3.6.1.7 above, close the open 8-inch valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

b. With one or more containment purge supply or exhaust isolation valves having a measured leakage rate in excess of the limits of the Containment Leakage Rate Testing Program, restore the inoperable valve(s) to OPERABLE status or isolate the affected penetration(s) so that the measured leakage rate does not exceed the limits of the Containment Leakage Rate Testing Program, within 24 hours and close the purge supply if the affected penetration is the exhaust penetration, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Each 8-inch containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is in accordance with the Containment Leakage Rate Testing Program.

4.6.1.7.2 Each 8-inch containment purge supply and exhaust isolation valve shall be verified to be sealed closed or open in accordance with Specification 3.6.1.7 at least once per 31 days.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent Containment Spray Systems shall be OPERABLE with each Spray System capable of taking suction from the RWST* and automatically transferring suction to the containment sump.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one Containment Spray System inoperable, restore the inoperable Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable Spray System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Spray System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position;

b. By verifying, that on recirculation flow, each pump develops a differential pressure of greater than or equal to 262 psi when tested pursuant to Specification 4.0.5;

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

   1) Verifying that each automatic valve in the flow path actuates to its correct position on a Containment Pressure-Hi-3 test signal, and

   2) Verifying that each spray pump starts automatically on a Containment Pressure-Hi-3 test signal.

d. At least once per 10 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

*In MODE 4, when the Residual Heat Removal System is in operation, an OPERABLE flow path is one that is capable of taking suction from the refueling water storage tank upon being manually realigned.

CONTAINMENT ENCLOSURE BUILDING

CONTAINMENT ENCLOSURE BUILDING STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.5.3 The structural integrity of the containment enclosure building shall be maintained at a level consistent with the Containment Leakage Rate Testing Program.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the structural integrity of the containment enclosure building not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.3 The structural integrity of the containment enclosure building shall be determined in accordance with the Containment Leakage Rate Testing Program. Any abnormal degradation of the containment enclosure building detected during the above required inspections shall be reported to the Commission in a Special Report pursuant to Specification 6.8.2 within 15 days.

6.14 MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID RADWASTE TREATMENT SYSTEMS*

6.14.1 Licensee-initiated major changes to the Radwaste Treatment Systems (liquid, gaseous, and solid):

a. Shall be reported to the Commission in the Annual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the SORC. The discussion of each change shall contain:

1) A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

2) Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

3) A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;

4) An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the License application and amendments thereto;

5) An evaluation of the change, which shows the expected maximum exposures to a MEMBER OF THE PUBLIC in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the License application and amendments thereto;

6) A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the change is to be made;

7) An estimate of the exposure to plant operating personnel as a result of the change; and

8) Documentation of the fact that the change was reviewed and found acceptable by the SORC.

b. Shall become effective upon review and acceptance by the SORC.

*Licensees may choose to submit the information called for in this Specification as part of the annual FSAR update.

6.15 CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program, dated September 1995," as modified by approved exceptions.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 49.6 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.15% of primary containment air weight per day.

The provisions of SR 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests.

Overall air lock leakage rate acceptance criterion is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

Each containment 8-inch purge supply and exhaust isolation valve leakage rate acceptance criterion is $\leq 0.01\ L_a$ when tested at P_a.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guidelines of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

Primary containment OPERABILITY is maintained by limiting leakage to $\leq 1.0\ L_a$, except prior to the first startup after performing a required Primary Containment Leakage Rate Testing Program leakage test. At this time, applicable leakage limits must be met. The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to $0.75\ L_a$ during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates is in accordance with the Containment Leakage Rate Testing Program.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the annulus atmosphere of 3.5 psi and (2) the containment peak pressure does not exceed the design pressure of 52 psig during LOCA conditions.

The maximum peak pressure expected to be obtained from a LOCA event is 49.6 psig. The limit of 16.2 psia for initial positive containment pressure will limit the total pressure to 49.6 psia which is less than the design pressure and is consistent with the safety analyses.

3/4.6.1 PRIMARY CONTAINMENT (Continued)

3/4.6.1.5 AIR TEMPERATURE

The limitation in containment average air temperature ensures that the containment average air temperature does not exceed the initial temperature condition assumed in the overall safety analysis for a steam line break accident. Measurements shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 52 psig in the event of a LOCA. A visual inspection in accordance with the Containment Leakage Rate Testing Program demonstrates this capability.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 36-inch containment purge supply and exhaust isolation valves are required to be sealed closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the Containment Purge System. To provide assurance that these containment valves cannot be inadvertently opened, the 36-inch containment shutdown purge supply and exhaust isolation valves are not utilized during operation on MODES 1-4, and a blind flange is installed establishing a Type "B" penetration. The penetration is surveilled in accordance with Surveillance Requirement 4.6.1.1a in MODES 1, 2, 3, and 4.

The use of the containment purge lines is restricted to the 8-inch purge supply and exhaust isolation valves since, unlike the 36-inch valves, the 8-inch valves are capable of closing during a LOCA or steam line break accident. Therefore, the SITE BOUNDARY dose guideline values of 10 CFR Part 100 would not be exceeded in the event of an accident during containment PURGING operation. The total time the containment purge (vent) system isolation valves may be open during MODES 1, 2, 3, and 4 in a calendar year is determined by the actual need for opening the valves for safety-related reasons; e.g., containment pressure control or the reduction of airborne radioactivity to facilitate personnel access for surveillance and maintenance activities.

Leakage integrity tests with a maximum allowable leakage rate for containment purge supply and exhaust supply valves will provide early indication of resilient material seal degradation and will allow opportunity for repair before gross leakage failures could develop. The containment purge supply and exhaust isolation valves are leakage rate tested in accordance with the Containment Leakage Rate Testing Program.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the Containment Spray System ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

The two independent Containment Spray Systems provide post-accident cooling of the containment atmosphere. The Containment Spray Systems also provide a mechanism for removing iodine from the containment atmosphere, and, therefore, the time requirements for restoring an inoperable Spray System to OPERABLE status have been maintained consistent with those assigned other inoperable ESF equipment.

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the Spray Additive System ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH volume and concentration ensure a Ph value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This Ph band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The contained solution volume limit includes an allowance for solution not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the safety analyses.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of General Design Criteria 54 through 57 of Appendix A to 10 CFR Part 50.

The opening of locked or sealed closed containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing an operator, who is in constant communication with control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment.

Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the expected hydrogen generation associated with: (1) zirconium-water reactions, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," March 1971.

The Hydrogen Mixing Systems are provided to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

3/4.6.5 CONTAINMENT ENCLOSURE BUILDING

3/4.6.5.1 CONTAINMENT ENCLOSURE EMERGENCY AIR CLEANUP SYSTEM

The OPERABILITY of the Containment Enclosure Emergency Air Cleanup System ensures that during LOCA conditions containment vessel leakage into the annulus, and radioactive materials leaking from engineered safety features equipment, from the electrical penetration areas, and from the mechanical penetration tunnel, will be filtered through the HEPA filters and charcoal adsorber trains prior to discharge to the atmosphere.

3/4.6.5.2 CONTAINMENT ENCLOSURE BUILDING INTEGRITY

CONTAINMENT ENCLOSURE BUILDING INTEGRITY ensures that the release of radioactive materials from the primary containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with operation of the Containment Enclosure Emergency Air Cleanup System, will limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR Part 100 during accident conditions.

3/4.6.5.3 CONTAINMENT ENCLOSURE BUILDING STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment enclosure building will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to provide: (1) protection for the steel vessel from external missiles, (2) radiation shielding in the event of a LOCA, and (3) an annulus surrounding the primary containment that can be maintained at a negative pressure during accident conditions. A visual inspection in accordance with the Containment Leakage Rate Testing Program, is sufficient to demonstrate this capability.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 49 TO FACILITY OPERATING LICENSE NO. NPF-86

NORTH ATLANTIC ENERGY SERVICE CORPORATION

SEABROOK STATION, UNIT NO. 1

DOCKET NO. 50-443

1.0 INTRODUCTION

On September 12, 1995, the U.S. Nuclear Regulatory Commission (NRC) approved issuance of a revision to 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors" which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The NRC added Option B "Performance-Based Requirements" to allow licensees to voluntarily replace the prescriptive testing requirements of 10 CFR Part 50, Appendix J, with testing requirements based on both overall leakage rate performance and the performance of individual components.

By application dated June 4, 1996, North Atlantic Energy Service Corporation (North Atlantic/the licensee) requested changes to the Technical Specifications (TS) for Seabrook Station, Unit No. 1 (Seabrook). The proposed changes would permit implementation of 10 CFR Part 50, Appendix J, Option B. North Atlantic has established a "Containment Leakage Rate Testing Program," and has proposed adding this program to the TS. The program references Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak Test Program," which specifies a method acceptable to the NRC for complying with Option B.

Supplemental information to the application for license amendment was submitted by letter dated January 8, 1997. The supplement provided information related to leak rate testing of the containment purge and vent valves and did not change the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

Compliance with 10 CFR Part 50, Appendix J, provides assurance that the primary containment, including those systems and components which penetrate the primary containment, do not exceed the allowable leakage rate specified in the TS and Bases. The allowable leakage rate is determined so that the leakage assumed in the safety analyses is not exceeded.

On February 4, 1992, the NRC published a notice in the Federal Register (57 FR 4166) discussing a planned initiative to begin eliminating requirements marginal to safety which impose a significant regulatory burden. Appendix J of 10 CFR Part 50 was considered for this initiative and the staff undertook a

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study of possible changes to this regulation. The study examined the previous performance history of domestic containments and examined the effect on risk of a revision to the requirements of Appendix J. The results of this study are reported in NUREG-1493, "Performance-Based Leak-Test Program."

Based on the results of this study, the staff developed a performance-based approach to containment leakage rate testing. On September 12, 1995, the NRC approved issuance of this revision to 10 CFR Part 50, Appendix J, which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The revision added Option B "Performance-Based Requirements" to Appendix J to allow licensees to voluntarily replace the prescriptive testing requirements of Appendix J with testing requirements based on both overall and individual component leakage rate performance.

Regulatory Guide 1.163, was developed as a method acceptable to the NRC staff for implementing Option B. This regulatory guide states that the Nuclear Energy Institute (NEI) guidance document NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" provides methods acceptable to the NRC staff for complying with Option B with four exceptions which are described therein.

Option B requires that the RG or other implementation document used by a licensee to develop a performance-based leakage rate testing program must be included, by general reference, in the plant TS. The licensee has referenced RG 1.163 in the Seabrook TS.

Regulatory Guide 1.163 specifies an extension in Type A test frequency to at least one test in 10 years based upon two consecutive successful tests. Type B tests may be extended up to a maximum interval of 10 years based upon completion of two consecutive successful tests and Type C tests may be extended up to 5 years based on two consecutive successful tests.

By letter dated October 20, 1995, NEI proposed TS to implement Option B. After some discussion, the staff and NEI agreed on final TS which were attached to a letter from C. Grimes (NRC) to D. Modeen (NEI) dated November 2, 1995. These TS are to serve as a model for licensees to develop plant specific TS in preparing amendment requests to implement Option B.

For a licensee to determine the performance of each component, factors that are indicative of or affect performance, such as an administrative leakage limit, must be established. The administrative limit is selected to be indicative of the potential onset of component degradation. Although these limits are subject to NRC inspection to assure that they are selected in a reasonable manner, they are not TS requirements. Failure to meet an administrative limit requires the licensee to return to the minimum value of the test interval.

Option B requires that the licensee maintain records to show that the criteria for Type A, B and C tests have been met. In addition, the licensee must

maintain comparisons of the performance of the overall containment system and the individual components to show that the test intervals are adequate. These records are subject to NRC inspection.

3.0 EVALUATION

North Atlantic proposes to establish a Containment Leakage Rate Testing Program and to add this program to the TS. The program would reference RG 1.163, which specifies a method acceptable to the NRC for complying with Option B. This requires a change to existing TS 1.7, 3/4.6.1, and 3/4.6.5, and the addition of a new Section 6.15 relating to the Containment Leakage Rate Testing Program. The corresponding bases were also modified.

Option B permits a licensee to choose Type A; or Type B and C; or Type A, B and C; testing to be done on a performance basis. North Atlantic has elected to perform Type A, B and C testing on a performance basis.

The TS changes proposed by North Atlantic are in compliance with the requirements of Option B and consistent with the guidance of RG 1.163, and the generic TS of the November 2, 1995, letter and are, therefore, acceptable to the staff.

The existing Seabrook TS 4.6.1.7.1 requires the containment purge supply and exhaust valves to be leak rate tested at least every 92 days to demonstrate that the leak rate does not exceed the specified limit. North Atlantic also has proposed to modify TS 3.6.1.7 and 4.6.1.7.1 by moving the specified leak rate acceptance criterion to TS 6.15 and adding reference to the Containment Leak Rate Testing Program. The valves then would be tested in accordance with RG 1.163 which specifies a test interval of 30 months. The current requirement to test these valves at least every 92 days is not an Appendix J requirement, but instead is based upon the findings of Generic Issue B-20, "Containment Leakage Due to Seal Degradation," that valves with resilient seals should be tested more frequently than required by Appendix J. The background for this conclusion is discussed in IE Circular 77-11, "Leakage of Containment Isolation Valves with Resilient Seals" (issued September 6, 1977).

By letter dated January 8, 1997, North Atlantic submitted the results of containment purge supply and exhaust valve leak rate testing and the maintenance performed on these valves from May 1986 to November 1995. Thirty-eight tests were performed on the containment purge supply valves, COP-V1 and COP-V2, and thirty-nine tests were performed on the containment purge exhaust valves, COP-V3 and COP-V4. None of the test leakage results exceeded the Technical Specification acceptance limit of 0.01 L_a, or 7.4 SCFH for these valves (where L_a is the maximum allowable containment leakage rate). The combined leakage rate for valves COP-V1 and COP-V2 has exhibited a small increasing trend over time, but the largest test leakage obtained to date has not exceeded 35 percent of the acceptance criterion. The combined leakage rate for valves COP-V3 and COP-V4 has exhibited a more stable rate, and the largest test leakage obtained to date has not exceeded 38 percent of the acceptance criterion. North Atlantic has not had to perform any maintenance on the valves that affect leaktightness of the valves.

Based on the testing results and maintenance history, the staff finds it acceptable to extend the maximum interval between leak rate tests on the containment purge supply and exhaust valves to 30 months as specified in RG 1.163.

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 surveillance requirements. 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 44359). 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.

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