North
Atlantic

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The Northeast Utilities System

August 6, 2001

Docket No. 50-443

NYN-01050

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Seabrook Station
License Amendment Request 01-05
"Administrative Changes To The Technical Specification Definitions"

North Atlantic Energy Service Corporation (North Atlantic) has enclosed herein License Amendment Request (LAR) 01-05. License Amendment Request 01-05 is submitted pursuant to the requirements of 10 CFR 50.90 and 10 CFR 50.4.

LAR 01-05 proposes changes to the Seabrook Station Technical Specifications (TS) Index, TS 1.0 ("Definitions") and TS Table 1.2.

In developing NUREG-1431 ("Standard Technical Specifications, Westinghouse Plants"), the NRC, in conjunction with the Westinghouse Owners Group (WOG), has developed standard definitions. The purpose of LAR 01-05 is to adopt many of these definitions. The enclosed Section I provides details of the changes. Those current definitions that are not being revised are either similar to NUREG-1431 wording or require further analysis to ascertain the impact to individual TSs.

The Station Operation Review Committee and the Nuclear Safety Audit Review Committee have reviewed LAR 01-05.

As discussed in the enclosed LAR Section IV, the proposed change does not involve a significant hazard consideration pursuant to 10 CFR 50.92. A copy of this letter and the enclosed LAR has been forwarded to the New Hampshire State Liaison Officer pursuant to 10 CFR 50.91(b). North Atlantic requests NRC Staff review of LAR 01-05, and issuance of a license amendment by January 10, 2002 (see Section V enclosed).

A001

North Atlantic has determined that LAR 01-05 meets the criterion of 10 CFR 51.22(c)(9) and 10 CFR 51.22(c)(10) for a categorical exclusion from the requirements for an Environmental Impact Statement (see Section VI enclosed).

Should you have any questions regarding this letter, please contact Mr. James M. Peschel, Manager - Regulatory Programs, at (603) 773-7194.

Very truly yours,
NORTH ATLANTIC ENERGY SERVICE CORP.

Ted C. Feigenbaum
Executive Vice President
and Chief Nuclear Officer

cc:

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North Atlantic

SEABROOK STATION UNIT 1

Facility Operating License NPF-86
Docket No. 50-443

License Amendment Request 01-05,
"Administrative Changes To The Technical Specification Definitions"

This License Amendment Request is submitted by North Atlantic Energy Service Corporation pursuant to 10CFR50.90. The following information is enclosed in support of this License Amendment Request:

- Section I - Introduction and Safety Assessment for Proposed Changes
- Section II - Markup of Proposed Changes
- Section III - Retype of Proposed Changes
- Section IV - Determination of Significant Hazards for Proposed Changes
- Section V - Proposed Schedule for License Amendment Issuance And Effectiveness
- Section VI - Environmental Impact Assessment

I, Ted C. Feigenbaum, Executive Vice President and Chief Nuclear Officer of North Atlantic Energy Service Corporation hereby affirm that the information and statements contained within this License Amendment Request are based on facts and circumstances which are true and accurate to the best of my knowledge and belief.

Ted C. Feigenbaum
Executive Vice President
and Chief Nuclear Officer

Sworn and Subscribed
before me this
6th day of August, 2001

Marilyn R. Sullivan
Notary Public

SECTION I

INTRODUCTION AND SAFETY ASSESSMENT FOR PROPOSED CHANGES

I. INTRODUCTION AND SAFETY ASSESSMENT OF PROPOSED CHANGES

A. Introduction

LAR 01-05 proposes changes to the Seabrook Station Technical Specifications (TS) Index, TS 1.0 ("Definitions") and TS Table 1.2.

In developing NUREG-1431 ("Standard Technical Specifications, Westinghouse Plants"), the NRC, in conjunction with the Westinghouse Owners Group (WOG), has developed standard definitions. The purpose of LAR 01-05 is to adopt many of these definitions. Those current definitions that are not being revised are either similar to NUREG-1431 wording or require further analysis to ascertain the impact to individual TSs.

The following definitions were modified to use the wording in Section 1.1 of NUREG-1431, Rev. 2:

ACTIONS
ACTUATION LOGIC TEST
CHANNEL CHECK
CORE ALTERATION
CORE OPERATING LIMITS REPORT (COLR)
DOSE EQUIVALENT I-131
ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME
MASTER RELAY TEST
MODE
OPERABLE - OPERABILITY
PHYSICS TESTS
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME
SHUTDOWN MARGIN (SDM)
SLAVE RELAY TEST
TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)

In addition, while the Seabrook Station Technical Specifications do not have a definition for a CHANNEL OPERATIONAL TEST (COT), it does have a definition for an ANALOG CHANNEL OPERATIONAL TEST (ACOT). Thus, the ACOT definition in the Seabrook Station Technical Specifications was modified to use the wording of the COT definition in NUREG-1431.

With respect to the definition of CHANNEL CALIBRATION, Seabrook Station has modified the definition in NUREG-1431. Below is the NUREG-1431 definition with ~~double strikeout~~ showing what was deleted and ***bold italic*** showing what was added by Seabrook Station:

> A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY. Calibration of instrument channels with ***sensors that are not adjustable, such as*** resistance temperature detectors (RTD*s*), ~~or~~ thermocouple sensors, ***or neutron detectors;*** may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps.

Please be aware that RTDs and thermocouples are not the only sensors that cannot be readily manipulated to provide the range of input signals needed to calibrate the channel. Neutron detectors; potential transformers, used in undervoltage sensing; and flow elements (e.g., orifices, annubars, and pitot tubes), such as those used to monitor auxiliary feedwater flow, also cannot be readily manipulated to provide a range of signals for channel calibration. Therefore, calibration of channels using these types of devices does not meet the verbatim definition of a CHANNEL CALIBRATION. Thus the NUREG-1431 CHANNEL CALIBRATION definition has given rise to exceptions like what occurs in NUREG-1431, Surveillance Requirement 3.3.1.11 (i.e., "Note Neutron detectors are excluded from CHANNEL CALIBRATION."). With the change delineated above, exceptions are no longer necessary.

Seabrook Station TS Table 1.2 was revised to reflect the presentation of MODES provided in Table 1.1-1 of NUREG-1431. In revising TS Table 1.2, the average reactor coolant temperature for MODES 1 and 2 are designated "NA" on the basis that temperature for these MODES is dictated by the minimum temperature for criticality and the operating program for T_{avg}. Currently TS Table 1.2 temperature requirement for MODE 1 and MODE 2 is $\geq$ 350°F. TS 3/4.1.1.4, "Minimum Temperature for Criticality," requires, in MODE 1 and MODE 2 that the temperature to be $\geq$ 551°F. In accordance with TS 3.0.4, prior to entering the Applicability of an LCO, the applicable surveillances must be met (in this case, verifying that reactor coolant temperature is $\geq$ 551°F). As a result, it is not necessary to explicitly include temperature requirements in the definition of MODES 1 and 2 in TS Table 1.2. In addition, the percent RATED THERMAL POWER "zero" values for MODES 3, 4, 5, and 6 are designated as "NA" on the basis that, with $k_{eff} < 0.99$, the THERMAL POWER would be zero.

The current TS Table 1.2 Footnote ** is revised to delete the phrases, "fuel in the reactor vessel," and "or with the head removed," consistent with NUREG-1431. The intent of applying the MODE definition only when fuel is in the reactor vessel has been moved to the definition of MODE - OPERATIONAL MODE. Since the reactor vessel head can only be removed if the head closure bolts are less than fully tensioned, there is no basis for including the phrase, "or with the head removed."

Finally, TS Table 1.2 is revised to clarify the head closure status and associated reactor coolant temperatures for plant conditions not previously satisfying a defined MODE, or satisfying more than one MODE. The phrase, "Required reactor vessel head closure bolts fully tensioned," is added as a footnote, which applies to the HOT SHUTDOWN and COLD SHUTDOWN MODE definitions. Clarification of these MODES with this footnote eliminates the overlap in defined MODES when the reactor coolant temperature is > 140°F. It is not the intent of the TS to allow an option of whether to apply LCOs applicable in the REFUELING MODE or to apply LCOs applicable in the HOT SHUTDOWN and/or COLD SHUTDOWN MODES. Additionally, the average coolant temperature and associated reactivity condition for the REFUELING MODE are replaced with the notation, "NA." The current definition of REFUELING (in TS Table 1.2) would cease to be applicable when average coolant temperature exceeded 140°F. By defining the REFUELING MODE as including plant conditions with no specific coolant temperature range, sufficiently conservative restrictions are applied by the applicable LCOs during all fueled conditions with the reactor vessel head closure bolts detensioned. The average coolant temperature in MODE 6 is adequately addressed by the decay heat removal requirements of TS 3/4.9.8.1, "Residual Heat Removal and Coolant Circulation — High Water Level," and TS 3/4.9.8.2, "Residual Heat Removal and Coolant Circulation - Low Water Level." The reactivity condition for this MODE is assured by TS 3/4.9.1, "Boron Concentration." The intent of these changes is to provide clarity and completeness in avoiding any misinterpretation.

The Seabrook Station definition of MODE was actually entitled: "OPERATIONAL MODE - MODE" and started out stating: "An OPERATIONAL MODE (i.e., MODE) . . . ." The definition is being changed to: "MODE - OPERATIONAL MODE" and the definition will start out saying: "A MODE (i.e., OPERATIONAL MODE) . . . ." Please note that the remainder of the definition follows the wording in NUREG-1431. In addition, the title of Table 1.1 is being changed from "OPERATIONAL MODES" to "MODES." These changes help delineate that the preferred nomenclature is "MODE" which is in alignment not only with NUREG-1431 but also industry practice and at the same time maintain the use of "OPERATIONAL MODE" for documentation consistency.

Several definitions already used the wording delineated in NUREG-1431, or are unique Seabrook Station definitions, but they did not have their acronym included in the title block. These definitions are:

AXIAL FLUX DIFFERENCE (AFD)
DIGITAL CHANNEL OPERATIONAL CHECK (DCOT)
QUADRANT POWER TILT RATIO (QPTR)
RATED THERMAL POWER (RTP)
OFFSITE DOSE CALCULATION MANUAL (ODCM)
PROCESS CONTROL PROGRAM (PCP)

While the IDENTIFIED LEAKAGE and the CONTROLLED LEAKAGE definitions have been modified to reflect NUREG-1431, Seabrook Station is not adopting the NUREG-1431 format for LEAKAGE, but is rather maintaining its separate definitions for: CONTROLLED LEAKAGE, IDENTIFIED LEAKAGE, PRESSURE BOUNDARY LEAKAGE, and UNIDENTIFIED LEAKAGE. These separate definitions are being maintained at this time since they already utilize the wording of NUREG-1431, and to change all the affected Specifications to meet this format, is not considered to be an effective use of Seabrook Station resources.

Finally there are changes, which include, deletion of the section numbers (i.e., 1.1 through 1.43), alphabetizing of the definitions, and updating the TS Index. These editorial changes also lead to the deletion of the "(NOT USED)" labeling.

B. Safety Assessment of Proposed Changes

NUREG-1431, Rev. 2 contains the improved Standard Technical Specifications (STS) for Westinghouse plants. This revision incorporates the cumulative changes to Revision 1, which was published in April 1995. The changes reflected in Revision 2 resulted from the experience gained from license amendment applications to convert to these improved STS or to adopt partial improvements to existing technical specifications. NUREG-1431, Rev. 2 is the result of extensive public technical meetings and discussions among the Nuclear Regulatory Commission (NRC) staff and various nuclear power plant licensees, Nuclear Steam Supply System (NSSS) Owners Groups, and the Nuclear Energy Institute (NEI). The improved STS were developed based on the criteria in the Final Commission Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, dated July 22, 1993 (58 FR 39132), which was subsequently codified by changes to Section 36 of Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.36) (60 FR 36953).

The proposed changes to the TS Index, TS 1.0 and TS Table 1.2 are administrative in nature and simply update the Seabrook Station Operating License to reflect the improved STS. The proposed changes do not affect nor modify the physical configuration of the facility or the manner in which it responds to normal, transient or accident conditions. Finally, while these changes may afford North Atlantic operational flexibility, the changes are an enhancement and do not affect plant safety.

North Atlantic concludes that based upon the above discussion as well as the Determination of Significant Hazards for Proposed Changes, presented in Section IV, that the proposed changes do not adversely affect or endanger the health or safety of the general public or involve a significant safety hazard.

SECTION II

MARKUP OF PROPOSED CHANGES

Refer to the attached markup of the proposed changes to the Technical Specifications. The attached markup reflects the currently issued revision of the Technical Specifications listed below. Pending Technical Specifications or Technical Specification changes issued subsequent to this submittal are not reflected in the enclosed markup.

The following Technical Specification changes are included in the attached markup:

Technical Specification	Title	Page
INDEX	INDEX	i
1.0	Definitions	1-1 through 1-7
Table 1.2	Modes	1-8

INDEX

1.0 DEFINITIONS

1.0 DEFINITIONS

1.0 DEFINITIONS

and Bases

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

ACTION (ACTIONS) actions to be taken (within specified completion times)

~~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 (ACOT)

all devices in the channel required for channel OPERABILITY. or actual

necessary

~~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.

required for channel OPERABILITY,

AXIAL FLUX DIFFERENCE (AFD)

~~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 (of the parameter that the channel monitors.) (necessary)

~~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. The CHANNEL CALIBRATION

CHANNEL CHECK (assessment, by observation,)

~~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.

Calibration of instrument channels with sensors which are not adjustable, such as resistance temperature detectors (RTDs), thermocouple sensors, or neutron detectors, may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel.

The ANALOG CHANNEL OPERATIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps.

Move "CONTAINMENT ENCLOSURE BUILDING INTEGRITY" to this location

DEFINITIONS

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~~ to, or leakoff from, the reactor coolant pump seals.

CORE ALTERATION

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

CORE OPERATING LIMITS REPORT (COLR)

~~1.10~~ The CORE OPERATING LIMITS REPORT ~~(COLR)~~ is the unit specific document that provides ~~core operating~~ cycle specific parameter 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 (DCOT)

~~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.

DOSE EQUIVALENT I-131

6 ~~1.12~~ DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) ~~(microCurie/gram)~~ ~~which~~ that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in NRC Regulatory Guide 1.109, Revision 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

6 ~~1.13~~ $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the sample) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the radionuclides in the sample with half-lives greater than 10 minutes.

ENGINEERED SAFETY FEATURES RESPONSE TIME

6 ~~1.14~~ The ENGINEERED SAFETY FEATURES (ESF) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ~~ESF~~ actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable. INSERT A

FREQUENCY NOTATION

6 ~~1.15~~ The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GASEOUS RADWASTE TREATMENT SYSTEM

6 ~~1.16~~ A GASEOUS RADWASTE TREATMENT SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting Reactor Coolant System offgases from the Reactor Coolant System and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

6 ~~1.17~~ IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to ~~a sump~~ a collection system, a sump, or a ~~or~~ collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of Leakage Detection Systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System.

INSERT A

The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

Consist of energizing all master relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required master relay.

MASTER RELAY TEST

~~1.18~~ A MASTER RELAY TEST shall ~~be the energization of each master relay and verification of OPERABILITY of each relay.~~ The MASTER RELAY TEST shall include a continuity check of each associated slave relay. The MASTER RELAY TEST may be performed by means of any series of sequential, overlapping, or total steps.

MEMBER(S) OF THE PUBLIC

~~1.19~~ MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

~~1.20~~ The OFFSITE DOSE CALCULATION MANUAL ~~(ODCM)~~ shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.7.6 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.8.1.3 and 6.8.1.4.

OPERABLE - OPERABILITY

normal or emergency — Safety — and — , and

~~1.21~~ A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling ~~or~~ seal water, lubrication ~~or~~ other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

specified safety

OPERATIONAL MODE - MODE

~~1.22~~ An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, ~~and~~ average reactor coolant ~~temperature~~ specified in Table 1.2 with fuel in the reactor vessel.

temperature, and reactor vessel head closure bolt tensioning

PHYSICS TESTS

~~1.23~~ PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation: (1) described in Chapter 14.0 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Nuclear Regulatory Commission.

PRESSURE BOUNDARY LEAKAGE

~~1.24~~ PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.

PROCESS CONTROL PROGRAM (PCP)

~~1.25~~ The PROCESS CONTROL PROGRAM ~~(PCP)~~ shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State Regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

~~1.26~~ PURGE or PURGING shall be any controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO (QPTR)

~~1.27~~ QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER (RTP)

~~1.28~~ RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3411 Mwt.

REACTOR TRIP SYSTEM (RTS) RESPONSE TIME

~~1.29~~ The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage. INSERT B

REPORTABLE EVENT

~~1.30~~ A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 of 10 CFR Part 50.

CONTAINMENT ENCLOSURE BUILDING INTEGRITY

~~1.31~~ CONTAINMENT ENCLOSURE BUILDING INTEGRITY shall exist when:

a. Each door in each access opening is closed except when the access opening is being used for normal transit entry and exit,

b. The Containment Enclosure Emergency Air Cleanup System is OPERABLE, and

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

Move in before "CONTAINMENT INTEGRITY"

INSERT B

The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

SHUTDOWN MARGIN (SDM)

~~1.32~~ SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition ~~assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.~~ INSERT C

SITE BOUNDARY

~~1.33~~ The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

SLAVE RELAY TEST

~~1.34~~ A SLAVE RELAY TEST shall INSERT D ~~be the energization of each slave relay and verification of OPERABILITY of each relay. The SLAVE RELAY TEST shall include a continuity check, as a minimum, of associated testable actuation devices.~~

~~1.35 (NOT USED)~~

SOURCE CHECK

~~1.36~~ A SOURCE CHECK shall be the qualitative assessement of channel response when the channel sensor is exposed to a source of increased radioactivity.

STAGGERED TEST BASIS

~~1.37~~ A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

THERMAL POWER

~~1.38~~ THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)

~~1.39~~ A TRIP ACTUATING DEVICE OPERATIONAL TEST shall consist of operating the Trip Actuating Device and verifying OPERABILITY ~~of alarm, interlock and/or trip functions. The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include adjustment, as necessary, of the Trip Actuating Device such that it actuates at the required Setpoint within the required accuracy.~~ INSERT E

INSERT C

assuming:

a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all RCCAs verified fully inserted by two independent means, it is not necessary to account for a stuck RCCA in the SDM calculation. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM, and

b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level.

INSERT D

consist of energizing all slave relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required slave relay. The SLAVE RELAY TEST shall include a continuity check of associated required testable actuation devices. The SLAVE RELAY TEST may be performed by means of any series of sequential, overlapping, or total steps.

INSERT E

of all devices in the channel required for trip actuating device OPERABILITY. The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the necessary accuracy. The TRIP ACTUATING DEVICE OPERATIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps.

UNIDENTIFIED LEAKAGE

~~1.40~~ UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

UNRESTRICTED AREA

~~1.41~~ An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

~~1.42 (NOT USED)~~

VENTING

~~1.43~~ VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

TABLE 1.1

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
N.A.	Not applicable.
P	Completed prior to each release.

TABLE 1.2

~~OPERATIONAL~~ MODES

MODE	REACTIVITY CONDITION, k_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	$\geq$ 0.99	> 5%	~~$\geq$ 350°F~~ N.A.
2. STARTUP	$\geq$ 0.99	$\leq$ 5%	~~$\geq$ 350°F~~ N.A.
3. HOT STANDBY	< 0.99	~~0~~ N.A.	$\geq$ 350°F
4. HOT ~~SHUTDOWN~~ SHUTDOWN**	< 0.99	~~0~~ N.A.	350°F > T_{avg} > 200°F
5. COLD ~~SHUTDOWN~~ SHUTDOWN**	< 0.99	~~0~~ N.A.	$\leq$ 200°F
6. ~~REFUELING**~~ REFUELING***	~~$\leq$ 0.95~~ N.A.	~~0~~ N.A.	~~$\leq$ 140°F~~ N.A.

*Excluding decay heat.

~~**Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.~~

**All reactor vessel head closure bolts fully tensioned.

***One or more reactor vessel head closure bolts less than fully tensioned.

SECTION III

RETYPE OF PROPOSED CHANGES

Refer to the attached retype of the proposed changes to the Technical Specifications. The attached retype reflects the currently issued version of the Technical Specifications. Pending Technical Specification changes or Technical Specification changes issued subsequent to this submittal are not reflected in the enclosed retype. The enclosed retype should be checked for continuity with Technical Specifications prior to issuance.

1.0 DEFINITIONS

1.0 DEFINITITIONS

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

ACTIONS

ACTIONS shall be that part of a Specification which prescribes required actions to be taken under designated conditions within specified completion times.

ACTUATION LOGIC TEST

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

ANALOG CHANNEL OPERATIONAL TEST (ACOT)

An ANALOG CHANNEL OPERATIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The ANALOG CHANNEL OPERATIONAL TEST shall include adjustments, as necessary, of the required alarm, interlock and/or Trip Setpoints required for channel OPERABILITY, such that the Setpoints are within the necessary range and accuracy. The ANALOG CHANNEL OPERATIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps.

AXIAL FLUX DIFFERENCE (AFD)

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

A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY. Calibration of instrument channels, with sensors which are not adjustable; such as resistance temperature detectors (RTDs), thermocouple sensors, or neutron detectors; may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. 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 ENCLOSURE BUILDING INTEGRITY

CONTAINMENT ENCLOSURE BUILDING INTEGRITY shall exist when:

a. Each door in each access opening is closed except when the access opening is being used for normal transit entry and exit,

b. The Containment Enclosure Emergency Air Cleanup System is OPERABLE, and

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

CONTAINMENT INTEGRITY

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

CONTROLLED LEAKAGE shall be that seal water flow supplied to, or leakoff from, the reactor coolant pump seals.

CORE ALTERATION

CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS REPORT (COLR)

The CORE OPERATING LIMITS REPORT is the unit specific document that provides cycle specific parameter limits for the current reload cycle. The cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 6.8.1.6. Plant operation within these limits is addressed in individual Specifications.

DIGITAL CHANNEL OPERATIONAL TEST (DCOT)

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.

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in NRC Regulatory Guide 1.109, Revision 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the sample) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the radionuclides in the sample with half-lives greater than 10 minutes.

ENGINEERED SAFETY FEATURES RESPONSE TIME

The ENGINEERED SAFETY FEATURES (ESF) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

FREQUENCY NOTATION

The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GASEOUS RADWASTE TREATMENT SYSTEM

A GASEOUS RADWASTE TREATMENT SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting Reactor Coolant System offgases from the Reactor Coolant System and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a collection system, a sump, or a collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of Leakage Detection Systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System.

MASTER RELAY TEST

A MASTER RELAY TEST shall consist of energizing all master relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required master relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay. The MASTER RELAY TEST may be performed by means of any series of sequential, overlapping, or total steps.

MEMBER(S) OF THE PUBLIC

MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

MODE - OPERATIONAL MODE

A MODE (i.e., OPERATIONAL MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.2 with fuel in the reactor vessel.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

The OFFSITE DOSE CALCULATION MANUAL shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.7.6 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.8.1.3 and 6.8.1.4.

OPERABLE - OPERABILITY

A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s), and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation: (1) described in Chapter 14.0 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Nuclear Regulatory Commission.

PRESSURE BOUNDARY LEAKAGE

PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.

PROCESS CONTROL PROGRAM (PCP)

The PROCESS CONTROL PROGRAM shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State Regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

PURGE or PURGING shall be any controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO (QPTR)

QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER (RTP)

RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3411 Mwt.

REACTOR TRIP SYSTEM (RTS) RESPONSE TIME

The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

REPORTABLE EVENT

A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 of 10 CFR Part 50.

SHUTDOWN MARGIN (SDM)

SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all RCCAs verified fully inserted by two independent means, it is not necessary to account for a stuck RCCA in the SDM calculation. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM, and

b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level.

SITE BOUNDARY

The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

SLAVE RELAY TEST

A SLAVE RELAY TEST shall consist of energizing all slave relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required slave relay. The SLAVE RELAY TEST shall include a continuity check of associated required testable actuation devices. The SLAVE RELAY TEST may be performed by means of any series of sequential, overlapping, or total steps.

SOURCE CHECK

A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

STAGGERED TEST BASIS

A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

THERMAL POWER

THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)

A TRIP ACTUATING DEVICE OPERATIONAL TEST shall consist of operating the Trip Actuating Device and verifying OPERABILITY of all devices in the channel required for trip actuating device OPERABILITY. The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the necessary accuracy. The TRIP ACTUATING DEVICE OPERATIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps.

UNIDENTIFIED LEAKAGE

UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

UNRESTRICTED AREA

An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

VENTING

VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

TABLE 1.1

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
N.A.	Not applicable.
P	Completed prior to each release.

TABLE 1.2

MODES

MODE	REACTIVITY CONDITION, k_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	≥ 0.99	$> 5\%$	N.A.
2. STARTUP	≥ 0.99	$\leq 5\%$	N.A.
3. HOT STANDBY	< 0.99	N.A.	$\geq 350°F$
4. HOT SHUTDOWN**	< 0.99	N.A.	$350°F > T_{avg} > 200°F$
5. COLD SHUTDOWN**	< 0.99	N.A.	$\leq 200°F$
6. REFUELING***	N.A.	N.A.	N.A.

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* Excluding decay heat.

** All reactor vessel head closure bolts fully tensioned.

*** One or more reactor vessel head closure bolts less than fully tensioned.

SECTION IV

DETERMINATION OF SIGNIFICANT HAZARDS FOR PROPOSED CHANGES

IV. DETERMINATION OF SIGNIFICANT HAZARDS FOR PROPOSED CHANGES

LAR 01-05 proposes changes to the Seabrook Station Technical Specifications (TS) Index, TS 1.0 ("Definitions") and TS Table 1.2.

In developing NUREG-1431 ("Standard Technical Specifications, Westinghouse Plants"), the NRC, in conjunction with the Westinghouse Owners Group (WOG), has developed standard definitions. The purpose of LAR 01-05 is to adopt many of these definitions.

NUREG-1431, Rev. 2 contains the improved Standard Technical Specifications (STS) for Westinghouse plants. This revision incorporates the cumulative changes to Revision 1, which was published in April 1995. The changes reflected in Revision 2 resulted from the experience gained from license amendment applications to convert to these improved STS or to adopt partial improvements to existing technical specifications. NUREG-1431, Rev. 2 is the result of extensive public technical meetings and discussions among the Nuclear Regulatory Commission (NRC) staff and various nuclear power plant licensees, Nuclear Steam Supply System (NSSS) Owners Groups, and the Nuclear Energy Institute (NEI). The improved STS were developed based on the criteria in the Final Commission Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, dated July 22, 1993 (58 FR 39132), which was subsequently codified by changes to Section 36 of Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.36) (60 FR 36953).

The NRC has encouraged licensees to upgrade their technical specifications consistent with those criteria and conforming, to the practical extent, to NUREG-1431, Rev. 2. The NRC requests that licensees adopting portions of the improved STS to existing technical specifications should adopt all related requirements, as applicable, to achieve a high degree of standardization and consistency.

In accordance with 10 CFR 50.92, North Atlantic has concluded that the proposed changes do not involve a significant hazards consideration (SHC). The basis for the conclusion that the proposed changes do not involve a SHC is as follows:

1. *The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.*

The proposed changes to TS Index, TS 1.0 and TS Table 1.2 do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, and configuration of the facility. In addition, the proposed changes do not affect the manner in which the plant responds in normal operation, transient or accident conditions. The proposed changes do not alter or prevent the ability of structures, systems and components (SSCs) to perform their intended function to mitigate the consequences of an initiating event within the acceptance limits assumed in the Updated Final Safety Analysis Report (UFSAR). Finally, while these changes may afford North Atlantic operational flexibility, the changes are an enhancement and do not affect plant safety.

The proposed changes do not affect the source term, containment isolation or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated in the Seabrook Station UFSAR. Further, the proposed changes do not increase the types and amounts of radioactive effluent that may be released offsite, nor significantly increase individual or cumulative occupational/public radiation exposures.

Therefore, it is concluded that these proposed revisions to TS Index, TS 1.0 and TS Table 1.2 do not involve a significant increase in the probability or consequence of an accident previously evaluated.

2. *The proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.*

This proposed changes to TS Index, TS 1.0 and TS Table 1.2 do not change the operation or the design basis of any plant system or component during normal or accident conditions. The proposed change incorporates definitions delineated in the improved Standard Technical Specifications (NUREG-1431). The proposed changes do not include any physical changes to the plant. In addition, the proposed changes do not change the function or operation of plant equipment or introduce any new failure mechanisms. The plant equipment will continue to respond per the design and analyses and there will not be a malfunction of a new or different type introduced by the proposed changes.

The proposed changes are administrative in nature and only correct, update and clarify the Seabrook Station Operating License to reflect the definitions in the improved Standard Technical Specifications. The proposed changes do not modify the facility nor do they affect the plant's response to normal, transient or accident conditions. The changes do not introduce a new mode of plant operation. While these changes may afford North Atlantic operational flexibility, the changes are an enhancement and do not affect plant safety. The plant's design and design basis are not revised and the current safety analyses remains in effect.

Thus, these proposed revisions to TS Index, TS 1.0 and TS Table 1.2 do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. *The proposed changes do not involve a significant reduction in the margin of safety.*

The proposed changes to TS Index, TS 1.0 and TS Table 1.2 are administrative in nature and only correct, update and clarify the Seabrook Station Operating License to reflect the improved Standard Technical Specifications. While these changes may afford North Atlantic operational flexibility, the changes are an enhancement and do not affect plant safety. The safety margins established through Limiting Conditions for Operation, Limiting Safety System Settings and Safety Limits as specified in the Technical Specifications are not revised nor is the plant design revised by the proposed changes.

Thus, it is concluded that these proposed revisions to TS Index, TS 1.0 and TS Table 1.2 do not involve a significant reduction in a margin of safety.

Based on the above evaluation, North Atlantic concludes that the proposed changes to TS Index, TS 1.0 and TS Table 1.2 do not constitute a significant hazard.

SECTIONS V AND VI

PROPOSED SCHEDULE FOR LICENSE AMENDMENT ISSUANCE AND EFFECTIVENESS

AND

ENVIRONMENTAL IMPACT ASSESSMENT

V. PROPOSED SCHEDULE FOR LICENSE AMENDMENT ISSUANCE AND EFFECTIVENESS

North Atlantic requests NRC review of License Amendment Request 01-05, and issuance of a license amendment by January 10, 2002, having immediate effectiveness and implementation within 90 days.

VI. ENVIRONMENTAL IMPACT ASSESSMENT

North Atlantic has reviewed the proposed license amendment against the criteria of 10 CFR 51.22 for environmental considerations. The proposed changes do not involve a significant hazards consideration, nor increase the types and amounts of effluent that may be released offsite, nor significantly increase individual or cumulative occupational radiation exposures. Based on the foregoing, North Atlantic concludes that the proposed changes meet the criterion delineated in 10 CFR 51.22(c)(9) and 10 CFR 51.22(c)(10) for a categorical exclusion from the requirements for an Environmental Impact Statement.