North Atlantic

North Atlantic Energy Service Corporation
P.O. Box 300
Seabrook, NH 03874
(603) 474-9521

The Northeast Utilities System

February 1, 2002

Docket No. 50-443

NYN-02006

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

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

References:

1) North Atlantic Letter to NRC (NYN-01015), License Amendment Request 01-05 "Administrative Changes to the Technical Specification Definitions" dated August 6, 2001

2) North Atlantic Letter to NRC (NYN-01083), "Supplemental Information for License Amendment Requests 01-04 and 01-05" dated November 2, 2001

North Atlantic Energy Service Corporation (North Atlantic) has enclosed herein a revision to License Amendment Request (LAR) 01-05. LAR 01-05 was originally forwarded to the Nuclear Regulatory Commission (NRC) by letter dated August 6, 2001 (Reference 1) and subsequently supplemented by letter dated November 2, 2001 (Reference 2). This revision to LAR 01-05 deletes changes to certain definitions originally proposed in LAR 01-05. This revision to LAR 01-05 simplifies the original request and limits the changes to Technical Specifications Sections 1.9 "Core Alteration," 1.14 "Engineered Safety Features Response Time," and 1.29 "Reactor Trip Response Time." Additionally, this revision to LAR 01-05 will narrow the scope of the definition section changes to only those needed to support refueling outage 08 activities. The definitions deleted from the original submittal of LAR 01-05 will be forwarded to the NRC in a future license amendment. The enclosed provides details of the revision to LAR 01-05 including a revised introduction, safety assessment, mark-up of the proposed changes and retype of the proposed changes.

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The revision to LAR 01-05 does not involve a significant hazard consideration pursuant to 10 CFR 50.92. An analysis was previously submitted to the NRC in the above referenced letters and concluded that no significant hazards are associated with this license amendment. The NRC reviewed the North Atlantic analysis and concluded that the three standards of 50.92(c) were satisfied and that the amendment request involved no significant hazards consideration (66 FR 59498, 59509) dated November 28, 2001. North Atlantic has determined that this proposed revision to LAR 01-05 does not affect the conclusions of the no significant hazards analysis previously submitted and evaluated by the NRC.

North Atlantic requests NRC review and approval of this revised LAR by April 19, 2002 to permit North Atlantic sufficient time to implement the changes to support refueling outage 08 activities in May 2002.

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:

H. J. Miller, NRC Regional Administrator
G. F. Wunder, NRC Project Manager, Project Directorate I-2
G. T. Dentel, NRC Senior Resident Inspector

STATE OF NEW HAMPSHIRE

Rockingham, ss. DATE 2/01/02

Then personally appeared before me, the above-named Ted C. Feigenbaum, being duly sworn, did state that he is the Executive Vice President and Chief Nuclear Officer of the North Atlantic Energy Service Corporation, that he is duly authorized to execute and file the foregoing information in the name and on the behalf of North Atlantic Energy Service Corporation and that the statements therein are true and accurate to the best of his knowledge and belief.

Marilyn R. Sullivan

Marilyn R. Sullivan, Notary Public
My Commission Expires: March 19, 2002

ENCLOSURE TO NYN-02006

SECTION I

INTRODUCTION AND SAFETY ASSESSMENT OF PROPOSED CHANGES

I. INTRODUCTION AND SAFETY ASSESSMENT OF PROPOSED CHANGES

A. Introduction

On August 6, 2001, North Atlantic Energy Service Corporation (North Atlantic) submitted License Amendment Request (LAR) 01-05 "Administrative Changes To The Technical Specification Definitions" by letter (NYN-01050) to the Nuclear Regulatory Commission (NRC) for review and approval. LAR 01-05 proposed administrative changes to the Technical Specifications (TS) Index, TS 1.0 "Definitions," and TS Table 1.2 "Operational Modes." The purpose of LAR 01-05 was to adopt many of the standard definitions outlined in NUREG 1431, "Standard Technical Specifications, Westinghouse Plants." Additionally, on November 2, 2001, North Atlantic provided an updated determination of significant hazards by letter (NYN-01083) to the NRC.

This revision to LAR 01-05 proposes changes to TS 1.9 "Core Alteration," 1.14 "Engineered Safety Features Response Time" and 1.29 "Reactor Trip System Response Time" only. The proposed revision to LAR 01-05 will narrow the scope of the LAR to only those definitions that North Atlantic has determined are necessary to support refueling outage 08 activities. This revision to LAR 01-05 adopts the corresponding definition outlined in Revision 2 of NRC NUREG-1431, "Standard Technical Specifications, Westinghouse Plants."

TS definition 1.9 "Core Alteration" is being changed to read as follows:

"CORE ALTERATION

1.9 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."

This definition change is administrative in nature and will more clearly identify the type of activities that constitute an actual core alteration to support station operation and refueling activities."

TS definition 1.14 "Engineered Safety Features Response Time" is being changed to read as follows:

"ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME

1.14 The 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."

TS definition 1.29 "Reactor Trip System Response Time" is being changed to read as follows:

"REACTOR TRIP SYSTEM (RTS) RESPONSE TIME

1.29 The RTS RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its RTS 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."

The changes to definitions to sections 1.14 and 1.29 are also administrative in nature but also support the Seabrook Station initiative outlined in LAR 01-07 "Changes to Certain Technical Specifications Associated With Response Time Testing." LAR 01-07 was forwarded to the NRC by letter (NYN-01105) dated December 21, 2001. The TS changes proposed in LAR 01-07 will provide North Atlantic operational flexibility by eliminating the periodic requirement for response time testing of certain components and systems.

B. Safety Assessment of Proposed Changes

There is no adverse safety impact as a result of the proposed changes to the definitions in sections TS 1.9 "Core Alteration," 1.14 "Engineered Safety Features Response Time" and 1.29 "Reactor Trip System Response Time." This revision to LAR 01-05 adopts the corresponding definition outlined in Revision 2 of NUREG-1431, "Standard Technical Specifications, Westinghouse Plants."

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 TS 1.9 "Core Alteration," 1.14 "Engineered Safety Features Response Time" and 1.29 "Reactor Trip System Response Time" are administrative in nature and simply update the Seabrook Station Operating License to reflect the improved STS. The change to TS

1.9 will also more clearly identify the type of activities that constitute an actual core alteration to support station operation and refueling activities. The changes to TS 1.14 and 1.29 will also support the changes submitted in LAR 01-07 to eliminate the periodic requirement for response time testing of certain components and systems. The safety assessment for the response time testing requirements of the subject components was provided to the NRC in LAR 01-07.

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, previously submitted to the NRC, 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
1.0	Definitions	1-2, 1-3, 1-5

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.

fuel, sources, or reactivity control components

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.

ALTERATIONS

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.

DOSE EQUIVALENT I-131

1.12 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which 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

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 FEATURE~~S~~ (ESF) RESPONSE TIME

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

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

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

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 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.

PROCESS CONTROL PROGRAM

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

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

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~~ RTS RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its ~~Trip Setpoint~~ RTS 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 0-rings) is OPERABLE.

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.

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.

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.

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
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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 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

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.

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

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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

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 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.

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.

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

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

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).

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.

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 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

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

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

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 RTS RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its RTS 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

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 0-rings) is OPERABLE.

SHUTDOWN MARGIN

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.

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 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 assessment 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

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.

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.