North Atlantic

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The Northeast Utilities System

June 20, 2000

Docket No. 50-443

NYN-00059

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

Seabrook Station
License Amendment Request 00-04,
"Reactor Coolant System Flow Measurement"

North Atlantic Energy Service Corporation (North Atlantic) has enclosed herein License Amendment Request (LAR) 00-04. LAR 00-04 is submitted pursuant to the requirements of 10CFR50.90 and 10CFR50.4.

LAR 00-04 proposes changes to the Seabrook Station Technical Specifications (TS) related to Reactor Coolant System (RCS) flow measurement surveillance requirement (SR) 4.2.5.3 contained in TS 3/4.2.5, "DNB Parameters, and the associated reactor trip function for Reactor Coolant Flow – Low, contained in TS Table 2.2-1, "Reactor Trip System Instrumentation Trip Setpoints."

The change to SR 4.2.5.3 would remove the prescriptive requirement for determining the RCS total flow rate by precision heat balance and incorporate a time limit for completion of the surveillance requirement.

Removal of the prescriptive requirement would allow RCS total flow rate to be determined within its limits by other flow monitoring/measurement methodology such as determination by elbow tap differential pressure (ΔP) measurement on the RCS cold legs.

Measurement of RCS total flow rate by use of elbow tap ΔP transmitters without the requirement to normalize to a precision RCS flow calorimetric each cycle has been evaluated by Westinghouse Electric Company LLC ("Westinghouse") for use at Seabrook Station. The evaluation justifies the use of elbow taps for RCS flow verification, as documented in WCAP-15404, "Justification of Elbow Taps for RCS Flow Verification at Seabrook Station," dated April 2000 (Proprietary). Use of elbow taps for RCS flow verification has been approved by the NRC for use at Joseph M. Farley Nuclear Power Plant[1], Units 1 and 2, and South Texas Project[2], Units 1 and 2.

1 Letter from L. Mark Padovan, NRC Project Manager Section 1, to Mr. D. N. Morey, Vice President - Farley Project, Southern Nuclear Operating Company, Inc., "Subject: Joseph M. Farley Nuclear Power Plant, Units 1 and 2- Re: Approval to Use WCAP-14750 at Farley (TAC Nos. M97325 and M97326), dated June, 30, 1999.

2 Issuance of License Amendments 108 and 95 for South Texas Project, Units 1 and 2, respectively, dated April 19, 1999.

APO1

In addition, with the proposed use of the elbow taps for RCS flow verification, changes to Functional Unit 12, Reactor Coolant Flow – Low reactor trip, in TS Table 2.2-1, "Reactor Trip System Instrumentation Trip Setpoints, " are necessary. The changes are to the Allowable Value, to reflect the allowed calibration tolerance of the protection racks due to instrument uncertainty calculations and noting that the Trip Setpoint is based on an indicated value rather than a measured value.

For historical tracking purposes, editorial changes have been made to the appropriate TS page to reflect the Amendments that materially changed the TS page.

Enclosed for NRC Staff review and approval are two copies of WCAP-15404 and two copies of the non-proprietary version, WCAP-15415, "Justification of Elbow Taps for RCS Flow Verification at Seabrook Station," dated April 2000. Also enclosed are a Westinghouse proprietary authorization letter, CAW-00-1397, accompanying affidavit, Proprietary Information Notice, Copyright Notice and letter NAH-00-028.

As WCAP-15404 contains information proprietary to Westinghouse, it is supported by an affidavit signed by Westinghouse, the owner of the information[3]. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.790 of the Commission's regulations.

Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.790 of the Commission's regulations.

The Station Operation Review Committee and the Nuclear Safety Audit Review Committee have reviewed LAR 00-04.

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 00-04, and issuance of a license amendment by October 15, 2000 (see Section V enclosed).

North Atlantic has determined that LAR 00-04 meets the criteria of 10 CFR 51.22(c)(9) 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

3 Correspondence with respect to the copyright or proprietary aspects of the aforementioned Westinghouse documents or the supporting Westinghouse Affidavit should reference CAW-00-1397 and should be addressed to H. A. Sepp, Manager of Regulatory and Licensing Engineering, Westinghouse Electric Company LLC ("Westinghouse"), P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

North Atlantic

SEABROOK STATION UNIT 1

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

License Amendment Request 00-04,
"Reactor Coolant System Flow Measurement"

North Atlantic Energy Service Corporation pursuant to 10CFR50.90 submits License Amendment Request 00-04. The following information is enclosed in support of this License Amendment Request:

- Section I - Introduction and Safety Assessment for Proposed Change
- Section II - Markup of Proposed Change
- Section III - Retype of Proposed Change
- Section IV - Determination of Significant Hazards for Proposed Change
- 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 License Amendment Request 00-04 are based on facts and circumstances which are true and accurate to the best of my knowledge and belief.

Sworn and Subscribed
before me this
20 day of June, 2000

Marilyn R. Sullivan
Notary Public

Ted Feigenbaum
Ted C. Feigenbaum
Executive Vice President and
Chief Nuclear Officer

Section I

Introduction and Safety Assessment for the Proposed Change

I. INTRODUCTION AND SAFETY ASSESSMENT OF PROPOSED CHANGES

A. Introduction

License Amendment Request (LAR) 00-04 proposes changes to the Seabrook Station Technical Specifications (TS) related to Reactor Coolant System (RCS) flow measurement surveillance requirement (SR) 4.2.5.3 contained in TS 3/4.2.5, "DNB Parameters, and the associated reactor trip function for Reactor Coolant Flow – Low, contained in TS Table 2.2-1, "Reactor Trip System Instrumentation Trip Setpoints."

The change to SR 4.2.5.3 would remove the prescriptive requirement for determining the RCS total flow rate by precision heat balance and incorporate a time limit for completion of the surveillance requirement.

Removal of the prescriptive requirement would allow RCS total flow rate to be determined within its limits by other flow monitoring/measurement methodology such as determination by elbow tap differential pressure (ΔP) measurement on the RCS cold legs.

Measurement of RCS total flow rate by use of elbow tap ΔP transmitters without the requirement to normalize to a precision RCS flow calorimetric each cycle has been evaluated by Westinghouse Electric Company (Westinghouse) for use at Seabrook Station. The evaluation justifies the use of elbow taps for RCS flow verification, as documented in WCAP-15404, "Justification of Elbow Taps for RCS Flow Verification at Seabrook Station. Use of elbow taps for RCS flow verification has been approved by the NRC for use at Joseph M. Farley Nuclear Power Plant, Units 1 and 2, and South Texas Project, Units 1 and 2.

In addition, with the proposed use of the elbow taps for RCS flow verification, changes to Functional Unit 12, Reactor Coolant Flow – Low reactor trip, in TS Table 2.2-1, "Reactor Trip System Instrumentation Trip Setpoints, " are necessary. The changes are to the Allowable Value, to reflect the allowed calibration tolerance of the protection racks due to instrument uncertainty calculations and noting that the Trip Setpoint is based on an indicated value rather than a measured value.

For historical tracking purposes, editorial changes have been made to the appropriate TS page to reflect the Amendments that materially changed the TS page.

B. Safety Assessment of Proposed Changes

With the introduction of lower neutron leakage cores the effects of temperature and flow gradients within the RCS hot legs have impacted the precision heat balance method of determining RCS flow rate that result in non-physical indications of flow decrease in the RCS. The elbow tap differential pressure (ΔP) indications will provide an indication of flow which is not dependent on future changes in hot leg measurement and therefore will provide an improvement in the flow surveillance accuracy.

Westinghouse was commissioned by North Atlantic to evaluate the option of using cold leg elbow tap flow measurements, based on the average of precision RCS flow calorimetrics and normalization of the elbow tap ΔP transmitters performed at Beginning of Cycles (BOC) 1 and 2, as an alternative to the currently used precision heat balance method of determining RCS total flow rate. Westinghouse's evaluation is presented in WCAP-15404, "Justification of Elbow Taps for RCS Flow Verification at The Seabrook Station". WCAP-15404 is the basis for LAR 00-04 proposed changes.

SR 4.2.5.3 currently requires the determination of the RCS total flow rate by precision heat balance measurement at an interval of at least once per 18 months. In the current method, a precision calorimetric heat balance measurement is taken on each side of the steam generator to calculate RCS flow rate based on gross steam generator thermal output divided by the enthalpy difference across the reactor vessel as measured by the RCS hot and cold leg resistance temperature detectors (RTDs).

The calculated RCS flow rates, as determined by the current surveillance requirement method, have markedly changed over the last seven fuel cycles at Seabrook Station with a trend approaching the Technical Specification minimum measured flow limit. These changes are not substantiated by the changes that have occurred in the system hydraulics, and are not confirmed by other indications of loop flow. This is readily apparent when compared with a trend of cold leg elbow tap indications and predicted changes due to known modifications which affect the system hydraulics, as determined by a hydraulic calculation method presented in WCAP-15404.

WCAP-15404 evaluated the Seabrook Station precision elbow tap ΔP data obtained during the pre-operational tests performed prior to BOC 1, and in the middle of cycle (MOC) 4, BOC 5, BOC 6, and BOC 7. Measurements from the control board or plant computer were also provided for BOC 2, BOC 3 and BOC 4, but were less accurate, so only the precision measurements were used to evaluate RCS flow changes. The evaluation accounted for reactor coolant pump impeller wear, steam generator tube plugging, removal of the RTD bypass system, and fuel design changes. This method of estimating/predicting RCS flow as well as the flow changes measured by the elbow taps are consistent and show very little flow decrease over the last seven fuel cycles; while the precision heat balance method shows considerably less RCS flow than that of the hydraulic calculation method. The difference is due to the overly conservative effects on the precision heat balance method from hot leg temperature streaming which is the cause of unsubstantiated RCS flow rate measurement decreases.

Hot leg temperature streaming phenomenon has been observed and reported by several Westinghouse PWRs. The phenomenon is a result of introducing the use of low neutron leakage cores in subsequent fuel cycles. The lower neutron leakage cores designs have a higher percentage of core power produced in the inner core regions. This leads to an increase temperature distribution (skewing) within the hot leg due to incomplete mixing and asymmetric flow patterns in the upper plenum that result in different temperature readings by one or more of the three hot leg RTDs. Because of the skewing of the temperature profile from the increased hot leg streaming, the three RTDs in each hot leg give an overstated indication of average hot leg temperature that does not accurately reflect true bulk average hot leg coolant temperature, resulting in an overly conservative temperature bias. Uncertainty in the hot leg temperature as indicated by the RTDs has been identified as the main contributor to calculated decreases in RCS flow rate since hot leg coolant density is calculated from the average T-hot temperature and impacts the calculated RCS flow rate when using the precision heat balance method. RCS calorimetric flow measurements are sensitive to temperature bias, since biases as small as 0.6 °F relative to the typical coolant ΔT of 60 °F results in an apparent change in calorimetric flow of 1 percent. Larger biases and lower flow measurements are likely to occur, as the core power gradients become larger and hot leg temperature gradients increase.

Due to the observed error when using the precision heat balance method of determining RCS flow rate (that is reliant on measurements provided by the hot leg RTDs which can indicate different temperatures in each loop, between loops, and can also change during the fuel cycle as the core power distribution changes) and the consequences related to core thermal margin and operating space, an alternative method of performing the flow surveillance using the cold leg elbow tap indications of flow is proposed. Cold leg elbow tap ΔP data is insensitive to changes in hot leg temperature gradients and sensitive to real mechanistic causes of changes in RCS flow, e.g., steam generator tube plugging, reactor coolant pump impeller smoothing.

The cold leg elbow tap configuration (3 per loop), installed in the crossover leg elbow in each loop and used to measure RCS flow rate, were not calibrated in a laboratory prior to the installation of the elbow in the RCS. The RCS consists of large diameter piping, which would make the calibration of the elbow taps in a laboratory expensive and difficult. Furthermore, elbow tap flow measurements are not accurate enough to define absolute flows in the RCS piping configuration due to the lack of the necessary straight lengths of upstream piping. Elbow taps are not significantly affected by phenomena such as changes in surface roughness (from crud deposits and fouling), elbow dimensional changes, erosion, and velocity distributions. The ΔP measurements are repeatable, as noted in WCAP-15404, and provide accurate indications of changes in flow.

At present, the flow limits for the elbow tap ΔP transmitters are determined based on the values determined by the precision calorimetric heat balance method at the beginning of each cycle. This normalization effectively changes the elbow tap coefficients each cycle. These adjustments give flow indication that are overly conservative since they are reflective of the calorimetric flow measurement that is biased due to the effects of hot leg temperature streaming. Therefore, to eliminate the effect of hot leg temperature streaming in the future determination of RCS flow, it is proposed that the average of previously performed BOC 1 and BOC 2 precision RCS flow calorimetrics be the basis for establishing baseline flow where the resulting effect of hot leg temperature streaming on baseline flow is minimal. The final normalized values for the elbow tap coefficients will be based on the average of BOC 1 and BOC 2 precision flow calorimetrics and that the elbow tap coefficients will no longer be adjusted to future calorimetrics. The RCS flow surveillance will then be performed by using future elbow tap ΔP indications. WCAP-15404 outlines a flow measurement procedure to determine RCS flow at the beginning of each fuel cycle. Application of the flow measurement procedure using normalized elbow tap measurements at Seabrook Station would result in the recovery of more than 3 percent flow, apparent loss attributed to changes in hot leg temperature streaming. The addition of intermediate flow mixing grids (IFMs) in upcoming Cycle 8 is expected to cause a real flow decrease of 0.3 to 0.4 percent. Thus, the recovery of flow margin is necessary to offset this decrease and to assure continued compliance with the RCS flow limiting condition for operation (LCO).

Using the new method of determining RCS flow rate requires the assurance that the most important potential safety need directly associated with RCS flow rate is maintenance of an adequate margin to prevent departure from nucleate boiling. The new method accounts for effect of instrument uncertainty on the determination of RCS flow and reactor protection, i.e., reactor trip due to low RCS flow rate, with the concerns being departure from nucleate boiling and an overtemperature condition.

The effects of not performing a normalization of the cold leg elbow taps for future cycles are evaluated in WCAP-15404. WCAP-15404 notes that use of cold leg elbow tap ΔP transmitters, re-calibrated to the average of the precision flow calorimetrics performed at BOC 1 and BOC 2 and no longer normalized at each refueling interval, requires inclusion of additional instrument uncertainties usually considered to be zeroed out by a normalization performed each cycle. The calculated instrument uncertainty is 2.3 percent flow (without accounting for feedwater venturi fouling). This uncertainty is slightly less than the current NRC licensed value of 2.4 percent Flow. Westinghouse determined that the instrument uncertainty calculation is reasonable with the Westinghouse approach approved by the NRC. The only significant difference is the assumption of normalization to previously performed RCS flow calorimetrics for Cycles 1 and 2. Based on these calculations, the minimum RCS flow that must be measured at 100 percent reactor thermal power (RTP) is maintained at 392,800 gpm. The calculated flow uncertainty is based on the following:

1) performance of a normalization of the elbow taps to the average of the RCS flow calorimetrics performed BOC 1 and BOC 2,

2) the current plant configuration,

3) inclusion of the effects of RTD Bypass Manifold Elimination, and

4) indication of RCS flow via the plant process computer or control board meters.

The evaluation concludes that the difference between the current Safety Analysis Limit (87% flow) and the nominal Trip Setpoint for Reactor Coolant Flow - Low (90% flow) specified in TS Table 2.2-1 for Functional Unit 12 is sufficient to allow for the increased instrument uncertainties due to the normalization, thus no changes to the nominal Trip Setpoint is required. In addition, the Allowable Value is changed to reflect the allowed drift for the process racks (in accordance with current NRC-approved Westinghouse instrument uncertainty methodologies, based on the 'as left' calibration tolerance defined in the uncertainty calculation for the trip). As a result of using the NRC-approved Westinghouse instrument uncertainty methodologies, the Total Allowance (TA), Z and Sensor Error (S) values are rendered meaningless, therefore these values are deleted. These changes are consistent with the five column to two column Reactor Trip Setpoint methodology reviewed and approved by the NRC in submittals from Millstone[4]. In addition, it is consistent with the format outlined in improved Standard Technical Specifications, NUREG-1431, Rev. 1. Since the time of reactor trip, as modeled in the various safety analyses is maintained, the conclusions of the safety analyses remain unchanged.

In addition, it is proposed to change the loop flow designation in the Reactor Coolant Flow – Low reactor trip function, TS Table 2.2-1, Functional Unit 12 from measured to indicated loop flow. This change is to be consistent with the wording recommended by Westinghouse. North Atlantic currently sets the low flow trip to 90 percent of measured flow regardless of what that measured flow is. This is no different than setting the low flow trip to 90 percent of indicated flow. Tripping on 90 percent of indicated flow for all loops will continue to ensure that the reactor trip setpoint remains conservative with respect to that modeled in the analysis. The associated Bases is changed accordingly.

The time limit associated with SR 4.2.5.3 could have been based on reaching 100 percent RTP, however, it is recognized that the plant could operate for an extended period at power levels between 90 to 100 percent RTP. Therefore, it was determined that 72 hours would be a reasonable limit to perform the RCS total flow rate verification when power exceeded 90 percent RTP.

Based on the foregoing assessment North Atlantic concludes that the Technical Specification changes proposed herein are acceptable. The changes allow RCS flow to be measured by using elbow tap ΔP transmitters without the requirement to normalize to a precision RCS flow calorimetric each cycle. The additional instrument uncertainties resulting from this process change have been accounted for and no change in the nominal Trip setpoint is required. The Allowable Value is changed to reflect the allowed calibration tolerance for the process racks. Therefore, the time of reactor trip, as modeled in the various safety analyses is maintained, and the conclusions of the safety analyses remain unchanged.

[4] Westinghouse letter NAH-00-028, "North Atlantic Energy Service Corporation Seabrook Unit 1 WCAPS 15404 and 15415 Appendix B," dated June 9, 2000.

Section II

Markup of the Proposed Changes

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

The following Technical Specifications are included in the attached markup:

Technical Specification	Title	Page(s)
Table 2.2-1	Reactor Trip System Instrumentation Trip Setpoints	2-5
Bases 2.2.1	Reactor Trip System Instrumentation Trip Setpoints - Reactor Coolant Flow	B 2-6
3/4.2.5	Power Distribution Limits – DNB Parameters	3/4 2-10

TABLE 2.2-1 (continued)
REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
11. Pressurizer Water Level - High	8.0	4.20	0.84	≤92% of instrument span	≤93.75% of instrument span
12. Reactor Coolant Flow - Low	N/A ~~2.5~~	N/A ~~1.9~~	N/A ~~0.6~~	≥90% of ~~measured~~ INDICATED loop flow	≥89.~~3~~6% of ~~measured~~ INDICATED loop flow
13. Steam Generator Water Level Low - Low	14.0	12.53	0.55	≥14.0% of narrow range instrument span	≥12.6% of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	15.0	1.39	0	≥10,200 volts	≥9,822 volts
15. Underfrequency - Reactor Coolant Pumps	2.9	0	0	≥55.5 Hz	≥55.3 Hz
16. Turbine Trip					
a. Low Fluid Oil Pressure	N.A.	N.A.	N.A.	≥500 psig	≥450 psig
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	≥1% open	≥1% open
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Pressurizer Pressure

In each of the pressurizer pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip, thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure that could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Setpoint trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer High Water Level trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power, the Pressurizer High Water Level trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full-power equivalent); and on increasing power, the Pressurizer High Water Level trip is automatically reinstated by P-7.

Reactor Coolant Flow

The Low Reactor Coolant Flow trips provide core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

INDICATED

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 90% of nominal full loop flow. Above P-8 (a power level of approximately 50% of RATED THERMAL POWER), an automatic Reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. Conversely, on decreasing power between P-8 and the P-7, an automatic Reactor trip will occur on low reactor coolant flow in more than one loop and below P-7 the trip function is automatically blocked.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified Setpoint provides allowances for starting delays of the Emergency Feedwater System.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

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

b. Pressurizer Pressure, ≥ 2185 psig*

c. Reactor Coolant System Flow shall be:

1. ≥ 382,800 gpm**; and,

2. ≥ 392,800 gpm***

APPLICABILITY: MODE 1.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

4.2.5.3 The RCS total flow rate shall be determined ~~by a precision heat balance measurement~~ to be within its limit WITHIN 72 HOURS OF EXCEEDING 90% ~~prior to operation above 98%~~ of RATED THERMAL POWER after each fuel loading. The provisions of Specification 4.0.4 are not applicable for entry into MODE 1.

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

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

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

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SECTION III

Retype of the Proposed Change

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 the Technical Specifications prior to issuance.

TABLE 2.2-1 (continued)
REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
11. Pressurizer Water Level - High	8.0	4.20	0.84	≤92% of instrument span	≤93.75% of instrument span
12. Reactor Coolant Flow - Low	N.A.	N.A.	N.A.	≥90% of indicated loop flow	≥89.6% of indicated loop flow
13. Steam Generator Water Level Low - Low	14.0	12.53	0.55	≥14.0% of narrow range instrument span	≥12.6% of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	15.0	1.39	0	≥10,200 volts	≥9,822 volts
15. Underfrequency - Reactor Coolant Pumps	2.9	0	0	≥55.5 Hz	≥55.3 Hz
16. Turbine Trip					
a. Low Fluid Oil Pressure	N.A.	N.A.	N.A.	≥500 psig	≥450 psig
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	≥1% open	≥1% open
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Pressurizer Pressure

In each of the pressurizer pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip, thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure that could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Setpoint trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer High Water Level trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power, the Pressurizer High Water Level trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full-power equivalent); and on increasing power, the Pressurizer High Water Level trip is automatically reinstated by P-7.

Reactor Coolant Flow

The Low Reactor Coolant Flow trips provide core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 90% of indicated loop flow. Above P-8 (a power level of approximately 50% of RATED THERMAL POWER), an automatic Reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. Conversely, on decreasing power between P-8 and the P-7, an automatic Reactor trip will occur on low reactor coolant flow in more than one loop and below P-7 the trip function is automatically blocked.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified Setpoint provides allowances for starting delays of the Emergency Feedwater System.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

a. Reactor Coolant System T_{avg}, $\leq$ 594.3°F

b. Pressurizer Pressure, $\geq$ 2185 psig*

c. Reactor Coolant System Flow shall be:

1. $\geq$ 382,800 gpm**; and,

2. $\geq$ 392,800 gpm***

APPLICABILITY: MODE 1.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

4.2.5.3 The RCS total flow rate shall be determined to be within its limit within 72 hours of exceeding 90% of RATED THERMAL POWER after each fuel loading. The provisions of Specification 4.0.4 are not applicable for entry into MODE 1.

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

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

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

Section IV

Determination of Significant Hazards for Proposed Change

IV. DETERMINATION OF SIGNIFICANT HAZARDS FOR PROPOSED CHANGE

License Amendment Request (LAR) 00-04 proposes changes to the Seabrook Station Technical Specifications (TS) related to Reactor Coolant System (RCS) flow measurement surveillance requirement (SR) 4.2.5.3 contained in TS 3/4.2.5, "DNB Parameters, and the associated reactor trip function for Reactor Coolant Flow – Low, contained in TS Table 2.2-1, "Reactor Trip System Instrumentation Trip Setpoints."

The change to SR 4.2.5.3 would remove the prescriptive requirement for determining the RCS total flow rate by precision heat balance and incorporate a time limit for completion of the surveillance requirement.

Removal of the prescriptive requirement would allow RCS total flow rate to be determined within its limits by other flow monitoring/measurement methodology such as determination by elbow tap differential pressure (ΔP) measurement on the RCS cold legs.

Measurement of RCS total flow rate by use of elbow tap ΔP transmitters without the requirement to normalize to a precision RCS flow calorimetric each cycle has been evaluated by Westinghouse Electric Company (Westinghouse) for use at Seabrook Station. The evaluation justifies the use of elbow taps for RCS flow verification, as documented in WCAP-15404, "Justification of Elbow Taps for RCS Flow Verification at Seabrook Station. Use of elbow taps for RCS flow verification has been approved by the NRC for use at Joseph M. Farley Nuclear Power Plant, Units 1 and 2, and South Texas Project, Units 1 and 2.

In addition, with the proposed use of the elbow taps for RCS flow verification, changes to Functional Unit 12, Reactor Coolant Flow – Low reactor trip, in TS Table 2.2-1, "Reactor Trip System Instrumentation Trip Setpoints, " are necessary. The changes are to the Allowable Value, to reflect the allowed calibration tolerance of the protection racks due to instrument uncertainty calculations and noting that the Trip Setpoint is based on an indicated value rather than a measured value.

For historical tracking purposes, editorial changes have been made to the appropriate TS page to reflect the Amendments that materially changed the TS page.

In accordance with 10 CFR 50.92, North Atlantic has reviewed the attached proposed changes and has concluded that the 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 do not adversely affect accident initiators or precursors nor alter the design, conditions, and configuration of the facility or the manner in which the plant is operated. 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).

Determination of RCS total flow rate by elbow tap ΔP measurement will not subject the reactor core to conditions adverse to nuclear safety. The proposed change does 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. The initial conditions for all accident scenarios modeled are the same. Therefore, the consequences of an accident occurring remain unchanged.

The evaluation for use of elbow tap ΔP measurement determined that sufficient margin exists to account for all reasonable instrument uncertainties, therefore no changes to installed equipment or hardware in the plant are required. Though the calibration process of the elbow tap ΔP transmitters has changed, i.e., normalization to previously performed precision RCS flow calorimetrics for Cycles 1 and 2 instead of normalization to a precision RCS flow calorimetric each cycle, this has been accounted for by the addition of instrument uncertainties usually considered to be zeroed out by normalization performed each cycle. Accounting for the additional instrument uncertainties yields a flow uncertainty that is slightly less (2.3 percent) than the current NRC licensed value (2.4 percent), thus no change is required to the nominal reactor trip setpoint for RCS flow. The proposed change has no adverse affect on component or system interactions. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes do not alter the design, conditions and configuration of the facility or the manner in which the plant is operated and maintained in a state of readiness. Existing system and component redundancy is not being changed by the proposed changes. Though the calibration process of the elbow tap ΔP transmitters has changed, i.e., normalization to previously performed precision RCS flow calorimetrics for Cycles 1 and 2 instead of normalization to a precision RCS flow calorimetric each cycle, this has been accounted for by the addition of instrument uncertainties usually considered to be zeroed out by normalization performed each cycle. Accounting for the additional instrument uncertainties yields a flow uncertainty that is slightly less than the current NRC licensed value, thus no change is required to the nominal reactor trip setpoint for RCS flow. The proposed change has no adverse affect on component or system interactions. The time of reactor trip remains the same. Therefore, since there are no changes to the design, conditions, configuration of the facility, or the manner in which the plant is operated and maintained in a state of readiness, the proposed changes do not create the possibility of a new or different kind of accident from any previously analyzed.

3. Involve a significant reduction in a margin of safety.

The proposed changes do not adversely affect equipment design or operation and there are no changes being made to the Technical Specification required safety limits or safety system settings that would adversely affect plant safety. The additional instrument uncertainties resulting from use of elbow tap ΔP transmitters without the requirement to normalize to a precision RCS flow calorimetric each cycle have been accounted for and no change in the nominal Trip Setpoint is required. The calculated instrument uncertainty is 2.3 percent flow. This uncertainty is slightly less than the current licensed value of 2.4 percent Flow. The time of reactor trip, as modeled in the various safety analyses, is maintained. Therefore, there is no significant reduction in a margin of safety.

Based on the above evaluation, North Atlantic concludes that the proposed changes do not constitute a significant hazard.

Sections V & 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 00-04, and issuance of a license amendment by October 15, 2000, having immediate effectiveness and implementation at commencement of Cycle 8 operation (presently scheduled mid-November 2000).

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 criteria delineated in 10 CFR 51.22(c)(9) for a categorical exclusion from the requirements for an Environmental Impact Statement.