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OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

MEMORANDUM TO: B. Paul Cotter, Jr.
Chief Administrative Judge
Atomic Safety and Licensing Board Panel

FROM: John C. Hoyle, Secretary

SUBJECT: HEARING REQUEST OF THE SEACOAST ANTI-POLLUTION LEAGUE

Attached is a request for hearing dated June 5, 1998, and submitted by Robert A. Backus on behalf of the Seacoast Anti-Pollution League (SAPL). The petition was filed in response to a notice of a proposed determination by the staff that the issuance of a license amendment to the North Atlantic Energy Service Corporation for Seabrook Station Unit No. 1 (Docket No.50-443) would involve no significant hazards considerations. The amendment would revise Technical Specifications on the frequency of steam generator inspections to accommodate a 24 month fuel cycle. The notice was published in the Federal Register at 63 Fed. Reg. 25101, 25113 (May 6, 1998) (copy attached).

The hearing request and related documents are being referred to you for appropriate action in accordance with 10 C.F.R. Sec. 2.772(j). Additionally, on June 11, 1998, Mr. Backus represented to an attorney in the Office of General Counsel that he intends to supplement SAPL's hearing request within a week.

Attachments: as stated

cc: Commission Legal Assistants
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CAA
OPA
EDO
NRR
Lillian M.Cuoco, Esquire
Northeast Nuclear Energy Company
Robert A. Backus, Esquire

SECY-EHD-008

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OF COUNSEL
ROBERT A. BACKUS
NANCY E. HART

June 5, 1998

Honorable Shirley Jackson
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Re: May 6, 1998 NAESCO License Exemption Request

Dear Chairman Jackson:

I am enclosing for your own review a copy of a letter making comments on behalf of the Seacoast Anti-Pollution League concerning the request of the operator of Seabrook Station to go to a 24 month fuel cycle.

I very much hope you will insure that we get a meaningful response to the objection to this request, which we believe raises very serious issues, giving the prior operating history at Seabrook, and the problem with steam generator tube degradation at many PWR's.

If the Commission decides not to institute a proceeding regarding this matter, I certainly hope you will direct the staff to hold a public hearing in the Seabrook area so the public will have a full opportunity to express concerns to the staff.

Very truly yours,

Robert A. Backus

RAB/acw

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June 5, 1998

Chief Rules and Directives Branch
Division of Administrative Services
Office Administration
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

ATTN: Secretary Hoyle

Re: May 6, 1998 NAESCO License Exemption Request

Dear Mr. Hoyle:

The purpose of this letter is to submit comments on a license exemption requested by North Atlantic Energy Service Corporation. The exemption request was published in Volume 62 of the Federal Register at page 25113 under date of May 6, 1998.

NAESCO, the requestor, is the operator of the Seabrook Nuclear Power Plant. The request seeks changes to the Technical Specifications to permit a 24 month refueling cycle at Seabrook. The staff, based upon the review of the licensee's application, has made a determination that the requested exemption involves no "significant hazards considerations."

This is to advise the Commission that the Seacoast Anti-Pollution League (SAPL), a concerned citizens organization, disagrees with the staff and believes that the Commission should either deny the exemption or institute a proceeding and grant a hearing on the exemption request. At the least, the Commission should afford the citizens in the Seabrook area an opportunity for a public hearing prior to granting the request.

SAPL's concerns about the exemption are based on four grounds: 1) the request will substantially lengthen the intervals between necessary surveillance of the steam generators; 2) the request will provide additional stress on and increase the likelihood of the fuel assembly degradation; 3) the exemption request will inevitably lead to the performance of more online maintenance, and 4) the exemption request may delay the discovery of either inadvertent or deliberate mispositioning of values or other components. Each of these factors can only result in an increase in the nuclear hazard and should therefore be held to involve a "significant hazards consideration."

For these reasons, discussed further below, SAPL believes the staff can not justify the granting of an exemption on the grounds that this action does not involve a significant hazards consideration.

1. Steam Generator Tube Degradation:

The staff, in recommending the exemption, discusses only the issue of less frequent steam generator surveillance, referencing Technical Specification 4.4.5.3. The staff states:

> "While the proposed changes will lengthen the intervals between surveillances, the increased interval has been evaluated; and based on the reviews of the steam generator tube Eddy Current Tests (ECT) inspections, it is concluded that the real growth rate of the only active degradation mechanism (Anti-Vibration Bar) (AVB) wear) identified to date that Seabrook Station is such that sufficient margin exists between the plugging criteria and structural limit such that no tubes are predicted to exceed the structural limit even with the longer surveillance interval."

Steam generator tube degradation is discussed, inter alia, in Inspection Report 97-03 which indicates, that, as of the date of the inspection, 36 tubes had been plugged. The report notes:

> "Although the number of tubes requiring plugs is low, the inspector recognized that the operating life is less than seven years. Most steam generated degradation problems have been found only after longer periods of operation. The E/C results to date indicate wall thinning attributable to flow induced vibratory relative motion between the tube and its intended support."

Based on the foregoing, it appears unreasonable for the staff to rely on the past growth rate of degradation due to AVB wear and then to boot strap from this alleged growth rate into a conclusion that extending the surveillance intervals by six months does not present a safety concern, since, as the staff has stated, major tube degradation may only develop after approximately seven years of operation. Seabrook began commercial operation in August, 1990.

SAPL, whose membership includes citizens of the State of Maine, is well aware of the rapid growth of steam generator tube degradation at the Maine Yankee plant and believes it is extremely unwise for the staff to conclude, with no supporting independent analysis, that increasing the interval for steam generator inspection at Seabrook by 25% is without safety significance.

SAPL is aware that extending the refueling intervals to 24 months is not in any way intended to enhance the safe operation of the plant, but only the economic viability of the plant on behalf of its utility owners, all of whom are facing competitive pressures. Given this circumstance, it is unacceptable for the staff to conclude that a major increase in the steam generator surveillance intervals, beyond that allowed by the current technical specifications, is acceptable.

2. Stress on Nuclear Fuel Cladding:

As the staff will be aware, at the time of original full power licensing, Seabrook was anticipated to have annual refuelings. Subsequently, the staff approved extending the refuelings to 18 months. If the present exemption is allowed, the refuelings will be double that anticipated when the plant went into operation.

It is SAPL's understanding that this increased operational period is achieved both by the use of more highly enriched fuel and an increase in the burn up of that fuel.

Both of these factors may cause additional stresses on fuel cladding, through the build up of gaseous by products near the end of the run. This potential has not been sufficiently evaluated by the Commission. The problem is addressed in a paper submitted by G. Rothwell and J. Russ "On the Optimal Life of Nuclear Power Plants." (1995). Rothwell and Russ acknowledge that "refueling durations are the most important factors limiting achievable availability factors." They add:

> "One of the difficult problems confronting nuclear plant operators is to determine the optimal length of operating (or refueling) cycles. There is a primary trade off between (1) the potential improvement and capacity factor with longer operating cycles and (2) the potential increased risk of unplanned mid-cycle outages due to fuel and other failures.... The high energy released by fission has deliterious effects on the structure of fuel rods. Some fission products appear as gasses that eventually create pressure within the fuel rods. As a result, a fuel rod can swell, crack, and become physically distorted to such an extent that it is no longer usable. The loss in fuel reactivity due to gradual depletion of radioactive uranium and build up of fission products, combined with the effect of radiation-induced fuel swelling and distortion, are limiting factors determining how long an NPP (Nuclear Power Plant) can run between refuelings. Maximum safe duration between refuelings is a function of the initial level of enrichment of the uranium, the design

of the fuel rods, and the fuel management strategy adopted by the operator."

With the 18 month fuel cycle currently in effect, Seabrook has already had fuel failure problems. As the result of detecting increases in noble gasses and iodine on December 10, 1996, it was determined that there were five failed fuel rods, in the first burned batch of Westhinghouse Vantage ZH Zurlo clad fuel assemblies.

Inspection Report 97-03 states, at p. 20:

> "The licensee root cause evaluation determined that a probable cause of the fuel failures was the combined effects of power history, core design and an operational strategy that resulted in interaction between the fuel pellets and the fuel cladding. The effective fuel assemblies apparently carried a very large load (produced high power) for all of the last cycle."

Since the staff has already concluded that the "power history" played a role in a fuel rod failure, on an 18 month cycle, it is inconceivable to SAPL how the staff can fail to assess, or give consideration to an increased risk, from extending that power history by 25% to two years.

SAPL calls on the Comission to demonstrate that these additional stresses, resulting from the longer operational run, will not result in a loss of the safety capability of the first barrier of defense against radioactive releases, the fuel assemblies themselves.

3. <u>Online Maintenance</u>:

SAPL is aware, but regrets, that pursuant to a letter of August 22, 1996, from Richard W. Cooper, II, Director of Division of Reactors Projects, the NRC staff authorized the use of online maintenance at Seabrook Station as of July 19, 1996. Online maintenance, by definition, involves the intentional disabling of safety related structures and components (SSC's) "that could initiate or effect a transient accident..." Reg. Guide 1.160, Introduction, June, 1993. SAPL would point out that Mr. Cooper's Letter of Authorization fails to mention, much less explain, the fact that this constitutes a complete reversal of the position the staff took on this very issue in 1987. In an Inspection Report (87-16, 10/21/87), the staff stated as follows:

> "Also, during this inspection period, the inspector confirmed with the station operations manager [New Hampshire Yankee, (the former Seabrook Station operator)] position that TS Limiting Condition for Operation (LCO) 3.0.0 is not intended for you as an operational convenience to permit redundant safety systems to be removed from service for a limited period of time. Based upon problems of interpretation of LCO 3.0.3 at other plants, <u>the NRC</u>

position is that voluntary entry into LCO 3.0.3 is unacceptable." (Emphasis added.)

SAPL has never been afforded an explanation of why the NRC changed its position from one that would not tolerate online maintenance, to one that permits online maintenance. Any claim that online maintenance is justified as a safety measure must be viewed with extreme scepticism given the obvious economic advantages of performing online maintenance, thereby shortening refueling outages, or now, under the proposed exemption, extending operational runs.

SAPL, in fact, believes that online maintenance is not properly authorized by 10 CFR 50.36(c)(2)(II). Nothing in the regulation authorizes voluntary, i.e., deliberate, disabling of the safety systems. This is documented by the fact that this requirement was part of the Commission's regulations prior to 1987, the time when the Commission's inspector advised Seabrook's former licensee that voluntary entry into the LCO's was not authorized. Furthermore, not one word on the regulatory analysis supporting the adoption of the Commission's maintenance rule, 10 CFR 50.65(A)(3), supports the use of online maintenance, and the environmental assessment fails to mention it.

NRC Inspection Manual 62706 illustrates methods for licensee compliance with the maintenance rule. This manual, which the staff cited when SAPL protested the use of online maintenance, states, at page 17C, "Assessment of Equipment Out of Service":

> "In order to minimize outage time and reduce costs, many licensees are increasing the amount of preventive maintenance being performed during power operation. This can result in the simultaneous removal of multiple systems from service, which can result in significant increases in risk during these periods. The NRC is concerned that some licensees may not be adequately analyzing the risk or safety impact associated with these unavailabilities. The failure to adequately evaluate safety when planning and scheduling maintenance has lead to simultaneous unavailabilities of multiple redundant or diverse systems at some sites, possibly leading to unacceptable increases in risk despite the fact that such configurations may not be prohibited by technical specifications. Technical specifications for most sites were crafted for random failure; voluntary removal of multiple systems from service may not be bounded by worst case single failure assumptions and technical specifications. The NRC is concerned that risk is significantly increased during periods when multiple redundance or diverse safety systems are unavailable due to preventive maintenance." (Emphasis added.)

This Manual clearly sets forth a concern about the improper use of online maintenance, which will be exacerbated if the proposed exemption is granted.

Mr. Cooper's letter of August 22, 1996, although authorizing online maintenance, acknowledged a "small risk associated with the unavailability" of certain safety systems due to online maintenance. No basis for assessing the risk to be small was provided, either in Mr. Cooper's letter, or by any of the regulatory analysis underlying the maintenance rule, nor is any basis provided for believing that "online maintenance can show a high degree of reliability that the equipment will perform its function if required," as the Cooper letter asserts.

Since, by definition, the systems taken deliberately out of service are important to safety, online maintenance represents an increase of the nuclear hazard which may not be offset by the claimed benefits.

The extension of the operational run to two years, before a refueling outage, obviously increases the need for online maintenance, increasing the very hazards that the NRC staff in the position taken in 1987 thought sufficiently serious to prohibit the practice. The exemption request provides no discussion of the increased risk that would be caused by the additional online maintenance required by the proposed exemption. Therefore, the exemption should not be deemed without safety significance.

4. <u>Inadequate Surveillance of Other Safety Items</u>:

In addition to the steam generators, the technical specifications indicate that the hydrogen recombiner system is to be subject to verification "at least once per 18 months during shutdown." A similar requirement exists for portions of the Containment Enclosure Emergency Air Clean-up System and the emergency diesel generators.[1] These items illustrate that a previously deemed necessary interval of surveillance, during shutdown, of 18 months for important systems is now no longer considered important to safety. SAPL protests this change of position, for which no rationale is offered.

In addition, SAPL is advised, and believes, that a refueling outage is the best opportunity for a licensee to find misaligned valves, either inadvertently or otherwise, or other evidence of tampering as well as numerous other conditions which may be important to safe operation. Nothing in the staff's proposed approval of the exemption addresses this aspect of increased risk.

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[1]SAPL is aware that under a previous exemption request, which, SAPL also protested (see letter to the Commission's Secretary from Mr. Steve Haberman of May 22, 1998), that NAESCO has requested a waiver of the current required surveillance frequency for the emergency diesel generators.

CONCLUSION

SAPL strongly protests the staff's preliminary conclusion that the licensee's request to extend Seabrook's run to two years between refuelings does not involve a significant hazards consideration. The staff has failed to evaluate many of the risks involved, and failed to properly justify its conclusion for the one risk it discusses, less frequent steam generators tube inspections.

In addition, the staff fails to acknowledge that, according to the last SALP report, performance at Seabrook is declining. As noted in Inspection Report 97-08, April 1, 1998, "Failure to correct these [3] conditions sooner indicates the decline in your performance with respect to analysis of root cause of problems as well as implementation of appropriate corrective action. This concern was previously highlighted in my January 23, 1998 letter transmitting the latest SALP report to you." (p.2.) A plant recently cited for four violations and considered to be in a state of declining performance should not be given the benefit of a 25% increase in its operational run without clear justification..

SAPL notes, finally, that NAESCO is a wholly owned subsidiary of Northeast Utilities, which through another wholly owned subsidiary, permitted the disastrous decline in the three Millstone Units, which has proved to be both costly for Northeast Utilities and embarrassing for the NRC. To suggest that the fourth, and currently only operating NU plant, should be given a "bonus" of permitting extended operation, with unresolved safety issues as a result, is unjustifiable.

We call on the Commission to reject the exemption request or, in the alternative, direct the institution of a proceeding under the Atomic Energy Act. We also request an opportunity to meet with the Commission concerning this issue.

Respectfully submitted,

Robert A. Backus

RAB/acw

cc: Governor Jeanne Shaheen
Congressman John Sununu
Senator Judd Gegg
Senator Bob Smith

MA 1498

North Atlantic

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

The Northeast Utilities System

April 8, 1998

Docket No. 50-443

NYN-98053

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Seabrook Station
License Amendment Request 98-03,
"Changes In Technical Specification Surveillance Intervals
To Accommodate A 24-Month Fuel Cycle Per Generic Letter 91-04"
Submittal No. 2

North Atlantic Energy Service Corporation (North Atlantic) has enclosed herein License Amendment Request (LAR) 98-03. LAR 98-03 is submitted pursuant to the requirements of 10CFR50.90 and 10CFR50.4.

LAR 98-03 is the second submittal in a planned series of License Amendment Requests which propose changes to the Seabrook Station Technical Specifications to accommodate fuel cycles of up to 24 months. The proposed changes are associated with surveillance requirements involving steam generator tube inspections that are currently performed at each 18-month or other specified outage interval. The License Amendment Request has been prepared in accordance with the generic guidance contained in NRC Generic Letter (GL) 91-04, "Changes In Technical Specification Surveillance Intervals To Accommodate A 24-Month Fuel Cycle."

The technical evaluation of the proposed increase in surveillance interval supports the conclusion that the effect on plant safety is insignificant. Analysis of historical surveillance data indicates tube degradation of the type experienced at Seabrook Station, Anti-Vibration Bar (AVB) Wear, will not reduce the margins of safety required by Regulatory Guide 1.121 for fuel cycles extended up to 24 months. In addition, the performance of the subject surveillances at the bounding surveillance interval of 24 months, plus 25% extension (30 months), does not invalidate assumptions in the plant licensing basis.

LAR 98-03 has been reviewed and approved by the Station Operation Review Committee and the Nuclear Safety Audit Review Committee.

As discussed in the enclosed LAR Section IV, the proposed changes do not involve a significant hazard consideration pursuant to 10CFR50.92. A copy of this letter and the enclosed LAR have been forwarded to the New Hampshire State Liaison Officer pursuant to 10CFR50.91(b). North Atlantic requests NRC review of LAR 98-03 and issuance of a license amendment by October 10, 1998 (see Section V enclosed).

North Atlantic has determined that LAR 98-03 meets the criteria of 10CFR51.22(c)(9) and 10CFR51.22(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. Terry L. Harpster, Director of Licensing Services, at (603) 773-7765.

Very truly yours,

NORTH ATLANTIC ENERGY SERVICE CORP.

Ted C. Feigenbaum

Ted C. Feigenbaum
Executive Vice President
and Chief Nuclear Officer

cc: H. J. Miller, NRC Regional Administrator
Craig W. Smith, NRC Project Manager, Project Directorate 1-3
R. K. Lorson, NRC Senior Resident Inspector

Mr. Woodbury P. Fogg, P.E., Director
New Hampshire Office of Emergency Management
State Office Park South
107 Pleasant Street
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North Atlantic

SEABROOK STATION UNIT 1

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

License Amendment Request No. 98-03,
"Changes in Technical Specification Surveillance Intervals To Accommodate A 24-Month Fuel Cycle Per Generic Letter 91-04"
Submittal No. 2

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 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

Sworn and Subscribed
before me this
8th day of April, 1998

Marilyn R. Sullivan
Notary Public

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

Section I

Introduction and Safety Assessment for the Proposed Changes

I. INTRODUCTION AND SAFETY ASSESSMENT OF PROPOSED CHANGES

A. Introduction

License Amendment Request (LAR 98-03) is the second submittal in a planned series of License Amendment Requests which propose changes to the Seabrook Station Technical Specifications to accommodate fuel cycles of up to 24 months for those selected surveillances that are currently performed at each 18-month or other outage interval.

The Technical Specifications proposed to be amended are:

4.4.5.3	Steam Generators - Inspection Frequencies
3.4.6.2c	Reactor Coolant System Leakage
3/4.4.5	Steam Generators Bases
3/4.4.6.2	Operational Leakage Bases

The proposed changes to the Seabrook Station Technical Specifications (TS) have been evaluated and modified in accordance with the generic guidance contained in NRC Generic Letter (GL) 91-04, "Changes In Technical Specification Surveillance Intervals To Accommodate A 24-Month Fuel Cycle." For the proposed changes contained herein, GL 91-04 requires that licensees evaluate the effect on safety of an increase in 18-month surveillance intervals to accommodate a 24-month fuel cycle. The evaluation should:

- support a conclusion that the effect on safety is small,
- confirm that historical plant maintenance and surveillance data support this conclusion and,
- confirm that assumptions in the plant licensing basis would not be invalidated on the basis of performing any surveillance at the bounding surveillance interval limit provided to accommodate a 24-month fuel cycle.

GL 91-04 further states that in consideration of these confirmations, the licensees need not quantify the effect of the change in surveillance intervals on the availability of individual systems or components.

Surveillance Requirement (SR) 4.4.5.3 is currently performed at intervals of not less than 12 nor more than 24 calendar months after the pre-service inspections. If two consecutive inspections, not including the preservice inspection, result in all inspection results falling into Category C-1 (as defined in T/S 4.4.5.2.) or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months. However, for plants having inspection results in the C-2 Category from inspections of steam generators (SGs) during either of the two previous inspections, the bounding interval for the next inspection would be 24 months from the last inspection.

A 24-month inspection interval may not always coincide with the next refueling outage when operating on fuel cycles of up to 24 months, particularly if any outage time is accumulated over the duration of the fuel cycle or if startup for the next fuel cycle is delayed following the completion of a SG inspection. Therefore, the proposed changes to Surveillance Requirement 4.4.5.3 and Limiting Condition for Operation 3.4.6.2.c of the Seabrook Station Technical Specifications provide an alternative to compensate for any delay that could cause the interval for steam generator inspections to occur near the end of a 24-month fuel cycle but before the refueling outage. The alternative includes, (1) an increase in the sample size of tubes examined, (2) a suitable analysis of the integrity of steam generator tubes, if the inspection results are in a C-2 or a C-3 category, and (3) for plant operation beyond 24 months from the previous steam generator tube inspection when the results of either of the two previous inspections are in the C-2 category, the reactor coolant system leakage through any one steam generator shall not exceed 100 gallons per day.

The proposed changes will modify existing Surveillance Requirement 4.4.5.3.a to reflect the increase in sample size of tubes examined and the requirement of performing an engineering assessment of the steam generator tubes if the inspection results are in a C-2 or C-3 Category. Surveillance Requirement 4.4.5.3.b will be modified to reflect the interval of steam generator tube inspections of either 30 or 40 months depending on the results of the two consecutive inspections. Surveillance Requirement 4.4.5.3.d has been added to clarify that the provisions of Technical Specification Section 4.0.2 do not apply to the extended steam generator inspection interval because Technical Specification Section 4.4.5.3.a addresses those conditions under which the 24-month surveillance interval for steam generator tube inspections may be extended. These proposed changes to SR 4.4.5.3 are consistent with GL 91-04 suggested Technical Specification wording.

Bases Section 3/4.4.4.5, "Steam Generators," has been modified to reflect the intent of the engineering assessment for steam generator integrity addressed in Technical Specification Section 4.4.5.3.a. This addition addresses those conditions under which the 24-month surveillance interval for steam generator tube inspections may be extended. This proposed change to Bases 3/4.4.4.5 is consistent with GL-91-04 suggested Technical Specification wording.

The proposed change to Technical Specification Limiting Condition for Operation 3.4.6.2.c will add a more restrictive limit of Reactor Coolant System leakage through the steam generators (100 gallons per day for any steam generator) for the Category C-2 condition with steam generator tube inspections beyond 24 months. This proposed change to TS 3.4.6.2.c is consistent with GL 91-04 suggested Technical Specification wording. In addition, the associated Bases Section, 3/4.4.6.2, "Operational Leakage" has been modified to reflect the more restrictive limit being imposed for Reactor Coolant System leakage through steam generators, for steam generators with Category C-2 tube inspection results with inspection intervals beyond 24 months.

In summary, the proposed changes are consistent with the suggested changes specified in GL 91-04. The technical evaluation of the components surveilled by the TS surveillance requirements addressed herein conclude that the effect on plant safety by the proposed extension at the bounding surveillance interval of 30 months to be insignificant. Analysis of historical surveillance data indicates tube degradation of the type experienced at Seabrook Station, Anti-Vibration Bar (AVB) Wear, will not reduce the margins of safety required by Regulatory Guide 1.121 for fuel cycles extended up to 24 months. The proposed changes do not alter the intent or method by which the surveillances are conducted, do not involve any physical changes to the plant, do not alter the way any structure, system or component (SSC) functions, and do not modify the manner in which the plant is operated. As such, the proposed changes to extend the surveillance intervals will not degrade the ability of any SSC to perform its safety function. In addition, the performance of the referenced surveillances at the bounding surveillance interval of 30 months (24 months plus 25% extension) does not adversely affect nor invalidate assumptions in the plant licensing basis.

B. Safety Assessment of Proposed Changes

There are four steam generators in the Reactor Coolant System (RCS), one per loop. The function of the steam generators is to remove the heat from the reactor coolant system in order to produce high quality steam to drive the turbine generator.

The four Westinghouse Model F steam generators are original components designed to ASME Section III. Each steam generator contains 5626 Thermally Treated (TT), Inconel 600 U-tubes (SB-163), hydraulically expanded into the tubesheet at each end. The steam generator tubing is nominally 0.688" O.D. with a 0.040" wall thickness. The tube bundle is supported with a series of "V" shaped Anti-Vibration Bars (AVBs) in the U-bend region and eight stainless steel Tube Support Plates (TSPs), which includes the flow distribution baffle as TSP #1 in the straight leg regions.

The tubesheet is drilled on a square pitch with a 0.98" spacing. Each tube is identified by a row and column number. Rows are orientated parallel to the primary side head divider plate whereas the columns are perpendicular to the divider plate.

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained, since the tubes comprise a significant fraction of the surface area of the reactor coolant pressure boundary (RCPB). The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

As an adjunct to the inservice inspection program, operational limits are imposed by the Technical Specifications for steam generator primary to secondary leakage to ensure 1) that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break, and 2) that tube integrity is maintained in the event of a steam line rupture or under LOCA conditions.

A key component of inservice inspection of steam generator tubing is the use of eddy current testing (ECT) techniques in locating defect areas in the steam generator tubing and for assessing the overall condition of the tubing. Tubes with imperfections that exceed the minimum acceptable tube wall thickness and operational limit are removed from service by plugging techniques.

Throughout Seabrook Station's operating history, steam generator tube plugging has been primarily due to AVB wear. AVB wear is a degradation process of steam generator tubes due to mechanical rubbing of the steam generator tubes with the anti-vibration bars. This degradation process is caused by flow-induced vibration of the steam generator tubes.

The steam generators have been inspected as groups of two steam generators (RC-E-11A & RC-E-11D or RC-E-11B & RC-E-11C), except for the first refueling outage (OR01) where all four steam generators were inspected. During OR01, 30% of the total number of steam generator (RC-E-11A, RC-E-11B, RC-E-11C, & RC-E-11D) tubes were inspected. Subsequently, the inspection sample was increased to 40% of the total number of tubes (RC-E-11B & RC-E-11C or RC-E-11A & RC-E-11D depending on which outage) for the second refueling outage (OR02) and third refueling outage (OR03). During the fourth refueling outage (OR04), 100% of tubes which were suspected as having the potential of experiencing AVB wear (43% of the total for RC-E-11A & RC-E-11D) were inspected. During the fifth refueling outage (OR05), 100% of the total number of tubes for RC-E-11B & RC-E-11C were inspected.

The OR05 inspection identified AVB wear in regions of the steam generators which previously were thought as not having the potential of AVB wear, i.e., tubes contained in the first 24 rows or less, closest to the divider plate. Seven (7) tubes were required to be plugged due to AVB wear. This brings the total number of steam generator tubes plugged due to AVB wear to 24 tubes. No other forms of inservice tube degradation have been identified after 5 cycles of operation other than loose part wear.

Loose part wear is identified by eddy current testing and Foreign Object Search and Retrieval (FOSAR) activities. An indication of a possible loose part requires an assessment of the indication with resolutions such as tube plugging, part retrieval and/or engineering evaluation.

Observed Seabrook tube AVB wear rates were used to evaluate the proposed extended inspection intervals against an allowable 75% through-wall structural limit, as specified in Regulatory Guide 1.121, "Bases For Plugging Degraded PWR Steam Generator Tubes." The analysis developed several cases to project AVB flaw growth rate through future cycles. In each case evaluated, it has been determined that the allowable 75% through-wall structural limit would not be exceeded.

During the most recent outage (OR05, May / June 1997), steam generator eddy current inspection identified a total of 163 tubes (434 flaws) from steam generators B and C as containing AVB flaws. Steam generators A and D were inspected during 1995 (OR04), in which a total of 161 tubes (378 flaws) were identified containing AVB flaws. A comparison of these eddy current inspections with previous inspections (S/G A & D - OR02, 1992; S/G B & C - OR03, 1994) was completed to determine the AVB wear rates. The average flaw growth rate for each time period between inspections was determined (22%/882 EFPDs for S/G A & D, and 15%/942 EFPDs for S/G B & C). The larger AVB wear flaw growth rate, (22%/882 EFPDs for S/G A & D) was used to show in the analysis that Seabrook's steam generator tubes will not exceed the 75% throughwall structural limit due to AVB flaw growth.

Wear rate and structural analyses for another Model F steam generator have been performed. A comparison of Seabrook Station's steam generator AVB flaw progression rates to these analyses shows the Seabrook Station steam generator AVB flaw progression rates to be considerably less. The Seabrook Station analyses concluded that a structural limit of 75% throughwall can be assumed for AVB wear in Seabrook Station steam generator tubing based on Regulatory Guide 1.121 criteria.

Therefore, given the present AVB growth rate at Seabrook Station, the steam generator tube inspection schedule, and the projected fuel cycle length, the maximum projected AVB flaw depth will not exceed the 75% throughwall structural limit at Seabrook Station.

Technical Specification Bases 3/4.4.6.2, Operational Leakage, states that the total steam generator tube leakage limit of 1 gpm for all steam generators not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a steam line rupture or under LOCA conditions.

To provide additional margin to accommodate a tube flaw which might grow at a greater than expected rate, for steam generators with Category C-2 tube inspection results with inspection intervals beyond 24 months, a more restrictive operational leakage limit of 100 gallons per day per steam generator is being proposed to TS 3.4.6.2 and its associated Bases. The revised limit is intended to provide additional assurance that should a significant leak be experienced in service the plant will be shut down in a timely manner. Furthermore, the revised limit is consistent with GL 91-04 suggested changes.

Other Potential Degradation Mechanisms

As stated previously, the Seabrook Station steam generator inspections over the past five cycles of operation have not identified secondary side flaws or tube degradation mechanisms other than AVB wear and loose parts wear. However, as is the case with any steam generator, the potential exists that a previously non-existent damage mechanism could develop in the future, such as degradation mechanisms associated with secondary and primary water chemistry.

North Atlantic has demonstrated the ability to effectively control secondary chemistry to industry recognized standards and to pro-actively address potential future issues such as controlling steam generator tube fouling by chemistry means in an integrated and programmatic manner. As such, secondary side steam generator corrosion is not anticipated to be a significant issue for the foreseeable future (i.e., within the next 10 years).

The Seabrook Station steam generators are considered to be less susceptible to Primary Water Stress Corrosion Cracking (PWSCC) because they are fabricated with thermally treated (TT) Alloy 600 tubing. In addition, the U-bends of the lower 10 rows of tubes have been stress relieved. Based on current industry data, it is concluded that TT Alloy 600, in combination with lower row stress relief is expected to provide better resistance to PWSCC within such stressed areas as tube / tubesheet roll transitions and U-bend areas.

The steam generator inspection programs will continue to use appropriate non-destructive examination (NDE) techniques to effectively monitor for the potential development of steam generator secondary side tube flaws and degradation mechanisms and will continue to include the appropriate NDE techniques for PWSCC detection. This inspection program also addresses the requirements of USNRC Generic Letter 95-03, "Circumferential Cracking of Steam Generator Tubes," for detection of circumferential cracking.

During the November, 1995 (OR04) steam generator inspection, a sample of tubes using a probe qualified for circumferential crack detection per Appendix H of the EPRI Steam Generator Examination Guidelines revealed no indications of degradation.

In conclusion, the effect on plant safety by the proposed extension of Steam Generator tube inspections at the bounding surveillance interval of 30 months to support fuel cycles of up to 24 months has been determined to be insignificant. Analysis of historical surveillance data indicates tube degradation of the type experienced at Seabrook Station, i.e., AVB Wear, will not reduce the margins of safety required by Regulatory Guide 1.121 for fuel cycles extended up to 24 months. The proposed changes do not alter the intent or method by which the surveillances are conducted, do not involve physical changes to the plant, do not alter the way a structure, system or component (SSC) functions, and do not modify the manner in which the plant is operated. As such, the proposed changes to extend the surveillance intervals will not degrade the ability of a SSC to perform its safety function. In addition, the performance of the referenced surveillances at the bounding surveillance interval of 30 months (24 months plus 25% extension) does not adversely affect nor invalidate assumptions in the plant licensing basis.

Section II

Markup of Proposed Changes

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 Specifications are included in the attached markup:

Technical Specification	Title	Page(s)
4.4.5.3	Steam Generators - Inspection Frequencies	3/4 4-15
3.4.6.2c	Reactor Coolant System Leakage	3/4 4-21
3/4.4.5	Steam Generators Bases	B 3/4 4-2a
3/4.4.6.2	Operational Leakage Bases	B 3/4 4-4

SURVEILLANCE REQUIREMENTS

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

INSERT A

a. The first inservice inspection shall be performed after 6 Effective Full-Power Months but no later than restart after first refueling. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections, not including the preservice inspection, result in all inspection results falling in Category C-1 or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months;

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40-month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3a.; the interval may then be extended to a maximum of once per 30 OR 40 months~~, and~~ AS APPLICABLE;

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

1) Primary-to-secondary tubes leak (not including leaks originating from tube-to-tubesheet welds) in excess of the limits of Specification 3.4.6.2, or

2) A seismic occurrence greater than the Operating Basis Earthquake, or

3) A loss-of-coolant accident requiring actuation of the Engineered Safety Features, or

4) A main steam line or feedwater line break; and

d. INSERT B

REACTOR COOLANT SYSTEM LEAKAGE

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 1 gpm total reactor-to-secondary leakage through all steam generators and 500 gallons per day through any one steam generator,

INSERT C

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. 40 gpm CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 psig ± 20 psig, and

f. 0.5 gpm leakage per nominal inch of valve size up to a maximum of 5 gpm at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve.* 1 2

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

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*Test pressures less than 2235 psig but greater than 150 psig are allowed. Observed leakage shall be adjusted for the actual test pressure up to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one-half power.

Inserts for Proposed Wording Changes to Technical Specification Requirements
T/S 3.4.6.2.c, 4.4.5.3, Bases 3/4.4.5 & 3/4.4.6.2

INSERT
A

If 20 percent of the tubes were inspected and the results were in the C-1 Category or if 40 percent of the tubes were inspected and were in the C-2 Category during the previous inspection, the next inspection may be extended up to a maximum of 30 months in order to correspond with the next refueling outage if the results of the two previous inspections were not in the C-3 Category. However, if the results of either of the previous two inspections were in C-2 Category, an engineering assessment shall be performed before operation beyond 24 months and shall provide assurance that all tubes will retain adequate structural margins against burst throughout normal operating, transient, and accident conditions until the end of the fuel cycle or 30 months, which ever occurs first.

INSERT
B

d. The provisions of specification 4.0.2 do not apply for extending the frequency for performing inservice inspections as specified in Specifications 4.4.5.3a. and b.

INSERT
C

For plant operation beyond 24 months from the previous steam generator tube inspection when the results of either of the two previous inspections are in the C-2 Category as defined by Specification 4.4.5.2, the leakage through any one steam generator not isolated from the Reactor Coolant System shall not exceed 100 gallons per day, 3/4.4.4 RELIEF VALVES (Continued)

(2) No Surveillance Requirement (ACOT or TADOT) exists for verifying automatic operation.

(3) The required ACTION for an inoperable PORV(s) (closing the block valve) conflicts with any presumed requirement for automatic actuation.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation, so that corrective measures can be taken.

INSERT D
NEW PARAGRAPH

Inserts for Proposed Wording Changes to Technical Specification Requirements
T/S 3.4.6.2.c, 4.4.5.3, Bases 3/4.4.5 & 3/4.4.6.2
(continued)

INSERT
D

An engineering assessment of steam generator tube integrity will confirm that no undue risk is associated with plant operation beyond 24 months of the previous steam generator tube inspection. To provide this confirmation, the assessment would demonstrate that all tubes will retain adequate structural margins against burst during all normal operating, transient, and accident conditions until the end of the fuel cycle. This evaluation would include the following elements:

1. An assessment of the flaws found during the previous inspection of each steam generator.

2. An assessment of the maximum flaw size that can be expected before the end of the current fuel cycle or 30 months, whichever comes first, and the corresponding structural margins relative to the criteria of Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes."

3. An update of the assessment model, as appropriate, based on comparison of the predicted results of the steam generator tube integrity assessment with actual inspection results from previous inspections.

INSERT
E

For plant operation beyond 24 months from the previous steam generator tube inspection when the results of either of the two previous inspections are in the C-2 Category as defined by Specification 4.4.5.2, the more restrictive leakage through any one steam generator not isolated from the Reactor Coolant System of 100 gallons per day is intended to provide additional margin to accommodate a tube flaw which might grow at a greater than expected rate. The more restrictive limit provides additional assurance that should a significant leak be experienced in service the plant will be shut down in a timely manner.

REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

The total steam generator tube leakage limit of 1 gpm for all steam generators not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

INSERT E

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the Leakage Detection Systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 40 gpm with the modulating valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the safety analyses.

The specified allowed leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series check valve failure. It is apparent that when pressure isolation is provided by two in-series check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA that bypasses containment, these valves should be tested periodically to ensure low probability of gross failure.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valve is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

1302 91

SECTION III

Retype of Proposed Changes

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.

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full-Power Months but no later than restart after first refueling. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If 20 percent of the tubes were inspected and the results were in the C-1 Category or if 40 percent of the tubes were inspected and were in the C-2 Category during the previous inspection, the next inspection may be extended up to a maximum of 30 months in order to correspond with the next refueling outage if the results of the two previous inspections were not in the C-3 Category. However, if the results of either of the previous two inspections were in C-2 Category, an engineering assessment shall be performed before operation beyond 24 months and shall provide assurance that all tubes will retain adequate structural margins against burst throughout normal operating, transient, and accident conditions until the end of the fuel cycle or 30 months, whichever occurs first. If two consecutive inspections, not including the preservice inspection, result in all inspection results falling in Category C-1 or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months;

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40-month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3a.; the interval may then be extended to a maximum of once per 30 or 40 months, as applicable;

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

   1) Primary-to-secondary tubes leak (not including leaks originating from tube-to-tubesheet welds) in excess of the limits of Specification 3.4.6.2, or

   2) A seismic occurrence greater than the Operating Basis Earthquake, or

   3) A loss-of-coolant accident requiring actuation of the Engineered Safety Features, or

   4) A main steam line or feedwater line break; and

4.4.5.3 Inspection Frequencies (continued)

d. The provisions of specification 4.0.2 do not apply for extending the frequency for performing inservice inspections as specified in Specifications 4.4.5.3a. and b.

REACTOR COOLANT SYSTEM LEAKAGE

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 1 gpm total reactor-to-secondary leakage through all steam generators and 500 gallons per day through any one steam generator. For plant operation beyond 24 months from the previous steam generator tube inspection when the results of either of the two previous inspections are in the C-2 Category as defined by Specification 4.4.5.2, the leakage through any one steam generator not isolated from the Reactor Coolant System shall not exceed 100 gallons per day.

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. 40 gpm CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 psig ± 20 psig, and

f. 0.5 gpm leakage per nominal inch of valve size up to a maximum of 5 gpm at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve.*

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

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*Test pressures less than 2235 psig but greater than 150 psig are allowed. Observed leakage shall be adjusted for the actual test pressure up to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one-half power.

3/4.4.4 RELIEF VALVES (Continued)

(2) No Surveillance Requirement (ACOT or TADOT) exists for verifying automatic operation.

(3) The required ACTION for an inoperable PORV(s) (closing the block valve) conflicts with any presumed requirement for automatic actuation.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation, so that corrective measures can be taken.

An engineering assessment of steam generator tube integrity will confirm that no undue risk is associated with plant operation beyond 24 months of the previous steam generator tube inspection. To provide this confirmation, the assessment would demonstrate that all tubes will retain adequate structural margins against burst during all normal operating, transient, and accident conditions until the end of the fuel cycle. This evaluation would include the following elements:

1. An assessment of the flaws found during the previous inspection of each steam generator.

2. An assessment of the maximum flaw size that can be expected before the end of the current fuel cycle or 30 months, whichever comes first, and the corresponding structural margins relative to the criteria of Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes."

3. An update of the assessment model, as appropriate, based on comparison of the predicted results of the steam generator tube integrity assessment with actual inspection results from previous inspections.

REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

The total steam generator tube leakage limit of 1 gpm for all steam generators not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions. For plant operation beyond 24 months from the previous steam generator tube inspection when the results of either of the two previous inspections are in the C-2 Category as defined by Specification 4.4.5.2, the more restrictive leakage through any one steam generator not isolated from the Reactor Coolant System of 100 gallons per day is intended to provide additional margin to accommodate a tube flaw which might grow at a greater than expected rate. The more restrictive limit provides additional assurance that should a significant leak be experienced in service the plant will be shut down in a timely manner.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the Leakage Detection Systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 40 gpm with the modulating valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the safety analyses.

The specified allowed leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series check valve failure. It is apparent that when pressure isolation is provided by two in-series check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA that bypasses containment, these valves should be tested periodically to ensure low probability of gross failure.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valve is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

Section IV

Determination of Significant Hazards for Proposed Change

IV. DETERMINATION OF SIGNIFICANT HAZARDS FOR PROPOSED CHANGES

License Amendment Request (LAR) 98-03 is the second submittal in a planned series of License Amendment Requests which propose changes to the Seabrook Station Technical Specifications to accommodate fuel cycles of up to 24 months. The proposed changes are associated with steam generator tube inspection surveillance requirements that are currently performed at each 18-month or other outage interval. The License Amendment Request has been prepared in accordance with the generic guidance contained in NRC Generic Letter (GL) 91-04, "Changes In Technical Specification Surveillance Intervals To Accommodate A 24-Month Fuel Cycle."

The Technical Specifications proposed to be amended are:

4.4.5.3	Steam Generators - Inspection Frequencies
3.4.6.2c	Reactor Coolant System Leakage
3/4.4.5	Steam Generators Bases
3/4.4.6.2	Operational Leakage Bases

In accordance with 10 CFR 50.92, North Atlantic has reviewed the attached proposed changes and has concluded that they 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.**

   Extending Surveillance Requirement (SR) 4.4.5.3 to accommodate a 24 month cycle for inspection of steam generator tubes structural integrity, as well as, imposing a more restrictive Limiting Condition for Operation (TS 3.4.6.2.c) for reactor coolant system leakage through Category C-2 steam generators, will neither exacerbate nor significantly increase the probability or consequences of an accident previously evaluated in the Seabrook Station UFSAR.

   The proposed changes to SR 4.4.5.3 do not alter the intent or method by which the surveillances are conducted, do not involve physical changes to the plant, do not alter the way structures, systems or components (SSCs) function, and do not modify the manner in which the plant is operated.

   The proposed change to TS 3.4.6.2.c imposes more restrictive limits on plant operations due to RCS leakage through steam generators. The proposed change does not involve physical changes to the plant or alter the way a SSC functions.

   The proposed changes to SR 4.4.5.3 and TS 3.4.6.2.c, and their associated Bases, will not adversely affect the ability of the steam generators to perform their intended safety function. Furthermore, the proposed changes do not adversely affect the physical protective boundaries of the plant. The proposed changes do not affect accident initiators or precursors and do not alter the design assumptions, conditions, configuration of the facility or the manner in which the plant is operated. The proposed changes do not alter or prevent the ability of 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). The proposed changes are administrative in nature and do not change the level of programmatic controls or the procedural details associated with aforementioned surveillance requirements. While the proposed changes will lengthen the interval between surveillances, the increase in interval has been evaluated; and based on the reviews of the steam generator tube eddy current test (ECT) inspections, it is concluded that the wear growth rate of the only active degradation mechanism (Anti-Vibration Bar (AVB) wear) identified to date at Seabrook Station is such that sufficient margin exists between the plugging criteria and structural limit such that no tubes are predicted to exceed the structural limit even with the longer surveillance interval.

Since there are no changes to previous accident analyses, the radiological consequences associated with these analyses remain unchanged, therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. Therefore, the proposed changes will not significantly increase the probability or consequences of any previously analyzed accident.

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

The proposed changes to TS 3.4.6.2 and SR 4.4.5.3, and associated Bases, do not alter the design assumptions, conditions, configuration of the facility or the manner in which the plant is operated. There are no changes to the source term, containment isolation or radiological release assumptions used in evaluating the radiological consequences in the Seabrook Station UFSAR. Existing system and component redundancy is not being changed by the proposed changes. The proposed changes have no impact on component or system interactions. The proposed changes are administrative in nature and do not change the level of programmatic controls and procedural details associated with the aforementioned surveillance requirements. Therefore, since there are no changes to the design assumptions, conditions, configuration of the facility, or the manner in which the plant is operated and surveilled, the proposed changes do not create the possibility of a new or different kind of accident from any previously analyzed.

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

The proposed changes to the surveillance intervals for SR 4.4.5.3 is still consistent with the basis for the interval. The intent or method of performing the surveillances remains unchanged. The more restrictive limit for leakage through any one steam generator placed in Category C-2, as well as, the requirement to do an engineering assessment of steam generator tube integrity, provides additional margin of ensuring safe plant operation.

In addition, there is no adverse affect on 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 proposed changes are administrative in nature and do not change the level of programmatic controls and procedural details associated with the aforementioned surveillance requirements. While the proposed changes will lengthen the interval between surveillances, the increase in interval has been evaluated; and based on the reviews of the steam generator tube ECT inspections, it is concluded that the wear growth rate of the only active degradation mechanism (AVB wear) identified to date at Seabrook Station is such that sufficient margin exists between the plugging criteria and structural limit such that no tubes are predicted to exceed the structural limit even with the longer surveillance interval. Therefore, extension of the current surveillance intervals to accommodate a 24 month cycle will not significantly degrade the ability, the availability or the reliability of the steam generators to perform their intended safety function, thus, it is concluded that 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 98-03 and issuance of a license amendment by October 10, 1998, having immediate effectiveness and implementation required within 60 days.

VI. ENVIRONMENTAL IMPACT ASSESSMENT

North Atlantic has reviewed the proposed license amendment against the criteria of 10CFR51.22 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 change meets the criteria delineated in 10CFR51.22(c)(9) and 10CFR51.22(c)(10) for a categorical exclusion from the requirements for an Environmental Impact Statement.

Safeguards Policy and Procedures Letter 1–50, Revision 1, is not warranted.

Alternatives to the Proposed Action

The proposed action is to amend NRC Source Material License SUA–648, for reclamation of the Heap Leach Area, as requested by Umetco. Therefore, the principal alternatives available to NRC are to:

1. Approve the license amendment request as submitted; or
2. Amend the license with such additional conditions as are considered necessary or appropriate to protect public health and safety and the environment; or
3. Deny the amendment request.

Based on its review, the NRC staff has concluded that the environmental impacts associated with the proposed action do not warrant either the limiting of Umetco's future operations or the denial of the license amendment. Additionally, in the TER prepared for this action, the staff has reviewed the licensee's proposed action with respect to the criteria for reclamation, specified in 10 CFR Part 40, Appendix A, and has no basis for denial of the proposed action. Therefore, the staff considers that Alternative 1 is the appropriate alternative for selection.

Finding of No Significant Impact

The NRC staff has prepared an EA for the proposed renewal of NRC Source Material License SUA–648. On the basis of this assessment, the NRC staff has concluded that the environmental impacts that may result from the proposed action would not be significant, and therefore, preparation of an Environmental Impact Statement is not warranted.

The EA and other documents related to this proposed action are available for public inspection and copying at the NRC Public Document Room, in the Gelman Building, 2120 L Street N.W., Washington, DC 20555.

Notice of Opportunity for Hearing

The Commission hereby provides notice that this is a proceeding on an application for a licensing action falling within the scope of Subpart L, "Informal Hearing Procedures for Adjudications in Materials and Operators Licensing Proceedings," of the Commission's Rules of Practice for Domestic Licensing Proceedings and Issuance of Orders in 10 CFR Part 2 (54 FR 8269). Pursuant to § 2.1205(a), any person whose interest may be affected by this proceeding may file a request for a hearing. In accordance with § 2.1205(c), a request for a hearing must be filed within thirty (30) days from the date of publication of this Federal Register notice. The request for a hearing must be filed with the Office of the Secretary either:

(1) By delivery to the Rulemakings and Adjudications Staff of the Office of the Secretary at One White Flint North, 11555 Rockville Pike, Rockville, MD 20852; or

(2) By mail or telegram addressed to the Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Rulemakings and Adjudications Staff.

Each request for a hearing must also be served, by delivering it personally or by mail to:

(1) The applicant, Umetco Mineral Corporation, P.O. 1029, Grand Junction, CO 815[illegible]

(2) The NRC staff, by delivery to the Executive Director of Operations, One White Flint North, 11555 Rockville Pike, Rockville, MD 20852, or

(3) By mail addressed to the Executive Director for Operations, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

In addition to meeting other applicable requirements of 10 CFR Part 2 of the Commission's regulations, a request for a hearing filed by a person other than an applicant must describe in detail:

(1) The interest of the requestor in the proceeding;

(2) How that interest may be affected by the results of the proceeding, including the reasons why the requestor should be permitted a hearing, with particular reference to the factors set out in § 2.1205(g);

(3) The requestor's areas of concern about the licensing activity that is the subject matter of the proceeding; and

(4) The circumstances establishing that the request for a hearing is timely in accordance with § 2.1205(c).

Any hearing that is requested and granted will be held in accordance with the Commission's "Informal Hearing Procedures for Adjudications in Materials and Operator Licensing Proceedings" in 10 CFR Part 2, Subpart L.

Dated at Rockville, Maryland, this 30th day of April 1998.

For the Nuclear Regulatory Commission.

Joseph J. Holonich,

Chief, Uranium Recovery Branch, Division of Waste Management, Office of Nuclear Material Safety and Safeguards.

[FR Doc. 98–11980 Filed 5–5–98; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Biweekly Notice; Applications and Amendments to Facility Operating Licenses Involving No Significant Hazards Considerations

I. Background

Pursuant to Public Law 97–415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Pub. L. 97–415 revised section 189 of the Atomic Energy Act of 1954, as amended (the Act), to require the Commission to publish notice of any amendments issued, or proposed to be issued, under a new provision of section 189 of the Act. This provision grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from April 10 through April 24, 1998. The last biweekly notice was published on April 22, 1998 (63 FR 19964).

Notice of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing

The Commission has made a proposed determination that the following amendment requests involve no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The basis for this proposed determination for each amendment request is shown below.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period.

However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish in the **Federal Register** a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administration Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By June 5, 1998, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Proceedings" in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) The nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemakings and Adjudications Staff, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, and to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for a hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of factors specified in 10 CFR 2.714(a)(1)(i)–(v) and 2.714(d).

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room for the particular facility involved.

and the COLR are unchanged. Since any future changes to these details in the Bases or the COLR will be evaluated per the requirements of 10 CFR 50.59 or other applicable change control provisions, no reduction in a margin of safety will result. As such, these proposed changes do not involve a significant reduction in a margin of safety.

Based on the above discussion, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Auburn Memorial Library, 1810 Courthouse Avenue, Auburn, NE 68305

Attorney for licensee: Mr. John R. McPhail, Nebraska Public Power District, Post Office Box 499, Columbus, NE 68602–0499

NRC Project Director: John N. Hannon

North Atlantic Energy Service Corporation, Docket No. 50–443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire

Date of amendment request: April 8, 1998.

Description of amendment request: The proposed change would revise Technical Specifications (TSs) 4.4.5.3, Steam Generators—Inspection Frequencies, and 3.4.6.2.c, Reactor Coolant System (RCS) Leakage, and the associated bases to accommodate fuel cycles of up to 24 months with respect to the allowed time interval between steam generator inservice inspections.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

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

Extending Surveillance Requirement (SR) 4.4.5.3 to accommodate a 24 month cycle for inspection of steam generator tubes structural integrity, as well as, imposing a more restrictive Limiting Condition for Operation (TS 3.4.6.2.c) for reactor coolant system leakage through Category C–2 steam generators, will neither exacerbate nor significantly increase the probability or consequences of an accident previously evaluated in the Seabrook Station [updated final safety analysis report] UFSAR.

The proposed changes to SR 4.4.5.3 do not alter the intent or method by which the surveillances are conducted, do not involve physical changes to the plant, do not alter the way structures, systems or components (SSCs) function, and do not modify the manner in which the plant is operated.

The proposed change to TS 3.4.6.2.c imposes more restrictive limits on plant operations due to RCS leakage through steam generators. The proposed change does not involve physical changes to the plant or alter the way a SSC functions.

The proposed changes to SR 4.4.5.3 and TS 3.4.6.2.c, and their associated Bases, will not adversely affect the ability of the steam generators to perform their intended safety function. Furthermore, the proposed changes do not adversely affect the physical protective boundaries of the plant. The proposed changes do not affect accident initiators or precursors and do not alter the design assumptions, conditions, configuration of the facility or the manner in which the plant is operated. The proposed changes do not alter or prevent the ability of 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). The proposed changes are administrative in nature and do not change the level of programmatic controls or the procedural details associated with aforementioned surveillance requirements. While the proposed changes will lengthen the interval between surveillances, the increase in interval has been evaluated; and based on the reviews of the steam generator tube eddy current test (ECT) inspections, it is concluded that the wear growth rate of the only active degradation mechanism (Anti-Vibration Bar (AVB) wear) identified to date at Seabrook Station is such that sufficient margin exists between the plugging criteria and structural limit such that no tubes are predicted to exceed the structural limit even with the longer surveillance interval.

Since there are no changes to previous accident analyses, the radiological consequences associated with these analyses remain unchanged, therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. Therefore, the proposed changes will not significantly increase the probability or consequences of any previously analyzed accident.

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

The proposed changes to TS 3.4.6.2 and SR 4.4.5.3, and associated Bases, do not alter the design assumptions, conditions, configuration of the facility or the manner in which the plant is operated. There are no changes to the source term, containment isolation or radiological release assumptions used in evaluating the radiological consequences in the Seabrook Station UFSAR. Existing system and component redundancy is not being changed by the proposed changes. The proposed changes have no impact on component or system interactions. The proposed changes are administrative in nature and do not change the level of programmatic controls and procedural details associated with the aforementioned surveillance requirements. Therefore, since there are no changes to the design assumptions, conditions, configuration of the facility, or the manner in which the plant is operated and surveilled, the proposed changes do not create the possibility of a new or different kind of accident from any previously analyzed.

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

The proposed change ( ) to the surveillance intervals for SR 4.4.5.3 is still consistent with the basis for the interval. The intent or method of performing the surveillances remains unchanged. The more restrictive limit for leakage through any one steam generator placed in Category C–2, as well as, the requirement to do an engineering assessment of steam generator tube integrity, provides additional margin of ensuring safe plant operation.

In addition, there is no adverse affect on 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 proposed changes are administrative in nature and do not change the level of programmatic controls and procedural details associated with the aforementioned surveillance requirements. While the proposed changes will lengthen the interval between surveillances, the increase in interval has been evaluated; and based on the reviews of the steam generator tube ECT inspections, it is concluded that the wear growth rate of the only active degradation mechanism (AVB wear) identified to date at Seabrook Station is such that sufficient margin exists between the plugging criteria and structural limit such that no tubes are predicted to exceed the structural limit even with the longer surveillance interval. Therefore, extension of the current surveillance intervals to accommodate a 24 month cycle will not significantly degrade the ability, the availability or the reliability of the steam generators to perform their intended safety function, thus, it is concluded that there is no significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis, and based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Exeter Public Library, Founders Park, Exeter, NH 03833

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, PO Box 270, Hartford, CT 06141–0270

NRC Project Director: Cecil O. Thomas

Northeast Nuclear Energy Company, et al., Docket No. 50–336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of amendment request: April 6, 1998.

Description of amendment request: The proposed amendment will modify