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JUL 15 2010

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U.S. Nuclear Regulatory Commission
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Washington, DC 20555-0001

Salem Nuclear Generating Station, Unit No. 1 and Unit No. 2
Facility Operating License Nos. DPR-70 and DPR-75
NRC Docket Nos. 50-272 and 50-311

Subject: Response to NRC Request for Additional Information dated June 17, 2010, Related to Various Sections of the Salem Nuclear Generating Station, Units 1 and 2 License Renewal Application

Reference: Letter from Ms. Bennett Brady (USNRC) to Mr. Thomas Joyce (PSEG Nuclear, LLC) "REQUEST FOR ADDITIONAL INFORMATION FOR SALEM NUCLEAR GENERATING STATION UNITS 1 AND 2 LICENSE RENEWAL APPLICATION (TAC NOS ME1834, ME 1836)" dated June 17, 2010

In the referenced letter, the NRC requested additional information related to various sections of the Salem Nuclear Generating Station, Units 1 and 2 License Renewal Application. Enclosed is the response to this request for additional information.

This letter and its enclosure contain no regulatory commitments.

If you have any questions, please contact Mr. Ali Fakhar, PSEG Manager - License Renewal, at 856-339-1646.

A121
NRR

JUL 15 2010

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 7/15/10

Sincerely,

Paul J. Davison

Paul J. Davison
Vice President, Operations Support
PSEG Nuclear LLC

Enclosure: A. Responses to Request for Additional Information
B. Updates to Affected LRA Sections
C. Update to License Renewal Commitment List

cc: S. Collins, Regional Administrator – USNRC Region I
B. Brady, Project Manager, License Renewal – USNRC
R. Ennis, Project Manager - USNRC
NRC Senior Resident Inspector – Salem
P. Mulligan, Manager IV, NJBNE
L. Marabella, Corporate Commitment Tracking Coordinator
Howard Berrick, Salem Commitment Tracking Coordinator

Enclosure A

Responses to Request for Additional Information related to Various Sections of the Salem Nuclear Generating Station, Units 1 and 2, License Renewal Application (LRA)

RAI 3.1.2.2.12-01
RAI 3.3.2.2-01
RAI 3.3.2.2-02
RAI 3.3.2.12-01
RAI 3.4.1-01
RAI 3.5.2.2.1.7-01

RAI 3.1.2.2.12-01

Background:

10 CFR § 54.21(a)(3) requires that applicants for license renewal demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis (CLB) for the period of extended operation. Thus, the component and the aging effect must be clearly identified and related to the aging management program that the applicant is proposing to manage the aging effect.

Issue:

License renewal application (LRA) Table 3.1.2-3, "Reactor Vessel Internals," does not distinguish the aging effect discussed in LRA Section 3.1.2.2.12, "Cracking Due to Stress Corrosion Cracking and Irradiation-Assisted Stress Corrosion Cracking (IASCC)," from that in LRA Section 3.1.2.2.17, "Cracking Due to Stress Corrosion Cracking, Primary Water Stress Corrosion Cracking, and Irradiation-Assisted Stress Corrosion Cracking." Specifically, Table 3.1.2-3 does not reference the aging effect due to primary water stress corrosion cracking at all. This could cause problems in the future in identifying the right components under the aging effect of either LRA Section 3.1.2.2.12 or LRA Section 3.1.2.2.17 if the industry programs that the applicant is committed to will give different inspection and evaluation guidelines for managing the aging effects of LRA Sections 3.1.2.2.12 and 3.1.2.2.17.

Request:

Provide a revised LRA Table 3.1.2-3 to identify the aging effect discussed in LRA Section 3.1.2.2.17, or justify combining LRA Sections 3.1.2.2.12 and 3.1.2.2.17 under the table column title "Aging Effect Requiring Management" in LRA Table 3.1.2-3.

PSEG Response:

Salem LRA Table 3.1.2-3, "Reactor Vessel Internals", does not differentiate the aging effect and mechanism of "Cracking Due to Stress Corrosion Cracking, Primary Water Stress Corrosion Cracking (PWSCC), and Irradiation-Assisted Stress Corrosion Cracking (IASCC)", from the aging effect and mechanism "Cracking Due to Stress Corrosion Cracking and Irradiation-Assisted Stress Corrosion Cracking (IASCC)".

The Salem LRA project methodology and associated procedures defined the aging mechanism of Stress Corrosion Cracking (SCC) to include other forms of aging mechanisms for stress corrosion cracking, which included PWSCC. Therefore, the aging mechanism, PWSCC, was inherently combined with the aging effect and mechanism of "Cracking Due to Stress Corrosion Cracking and Irradiation-Assisted Stress Corrosion Cracking (IASCC)" listed in LRA Table 3.1.2-3 under the table column title "Aging Effect Requiring Management".

The aging mechanism of PWSCC was also inherently combined with the aging effect and mechanisms of "Cracking Due to Stress Corrosion Cracking and Irradiation-Assisted Stress Corrosion Cracking (IASCC)" as discussed in LRA Section 3.1.2.2.17.

The aging effect and mechanisms of "Cracking Due to Stress Corrosion Cracking and Irradiation-Assisted Stress Corrosion Cracking (IASCC)", which includes the aging mechanism of PWSCC, will be managed by the PWR Vessel Internals aging management program (Salem LRA Appendix B, Section B.2.1.7). The PWR Vessel Internals aging management program is the GALL recommended aging management program for the aging effect and mechanisms of "Cracking Due to Stress Corrosion Cracking and Irradiation-Assisted Stress Corrosion Cracking (IASCC)" and "Cracking Due to Stress Corrosion Cracking, Primary Water Stress Corrosion Cracking, and Irradiation-Assisted Stress Corrosion Cracking", therefore, Salem will manage the aging mechanism of PWSCC using the PWR Vessel Internals aging management program.

RAI 3.3.2.2-01

Background:

SRP-LR paragraph 3.3.2.2.4.1 addresses cracking due to stress corrosion cracking and cyclic loading for non-regenerative heat exchanger components exposed to treated borated water greater than 60°C. The Generic Aging Lessons Learned (GALL) Report recommends the Water Chemistry Program to manage cracking and a plant-specific program to verify the effectiveness of the Water Chemistry Program where an acceptable program would include radioactivity monitoring and eddy current testing of tubes.

Issue:

The staff noted the applicant relies on continuous monitoring for radioactivity on the shell side of the non-regenerative stainless steel heat exchangers to detect cracking due to stress corrosion cracking and cyclic loading. The staff noted that monitoring of radioactivity on the shell side will not detect cracking until cracking has progressed through the wall of the component whereas the GALL Report recommends eddy current examination which would detect cracking before leakage.

Request:

Justify how not using the recommended eddy current testing technique would detect cracking before leakage.

PSEG Response:

As described in the Salem LRA, Section 3.3.2.2.4.1, the Water Chemistry aging management program (Salem LRA Appendix B, Section B.2.1.2) is credited to manage aging effects for the stainless steel non-regenerative heat exchangers (i.e., Letdown, Excess Letdown, and Seal Water heat exchangers) exposed to treated borated water greater than 60°C (140°F). The Salem Water Chemistry aging management program provides for monitoring and controlling the chemical environments of the primary cycle systems to mitigate aging effects, including cracking of stainless steel components, and is consistent with the NUREG-1801 XI.M2 Water Chemistry Program.

Reactor coolant system leakage for the tube-side of these heat exchangers and radioactivity of the shell-side water of these heat exchangers are monitored when the heat exchangers are in service during normal operation by installed instrumentation. The reactor coolant system leakage surveillance determines the leakage for the entire reactor coolant system and the letdown and charging flow path with a plant shutdown acceptance criterion of one gallon per minute unidentified leakage. If the reactor coolant leak rate increases 0.1 gallons per minute above the weekly average, actions are initiated by the shift personnel to identify the source of the leakage.

There are radiation monitors in the Component Cooling System to monitor any significant leakage into the system from the non-regenerative heat exchangers. The Component Cooling System is a closed-loop system which prevents direct release of radioactivity to the environment. The radioactivity level of the Component Cooling System, which provides cooling

for these heat exchangers, is typically less than 1.5×10^{-7} µCi/ml, which is less than detectable with the test instrumentation. The radioactivity level is also trended, and any significant increase in radioactivity would require entry into the Corrective Action Program for investigation and resolution.

The LRA will be revised to add the One-Time Inspection aging management program to verify that the Water Chemistry aging management program activities are effectively mitigating cracking of non-regenerative heat exchanger components exposed to treated borated water greater than 60°C (140°F). The One-Time Inspection aging management program will be revised to include eddy current testing of stainless steel tubes in a non-regenerative heat exchanger normally exposed to treated borated water greater than 60°C (140°F). The Salem One-Time Inspection aging management program is consistent with the NUREG-1801 XI.M32 One-Time Inspection Program.

NUREG-1800, Section 3.3.2.2.4.1, recommends that the effectiveness of the water chemistry control program be verified to ensure cracking is not occurring in stainless steel heat exchanger components exposed to treated borated water greater than 60°C (140°F). The monitoring of reactor coolant leakage on the tube-side, radioactivity monitoring of the shell-side water, and the One-Time Inspection aging management program activity to perform eddy current testing of stainless steel tubes in a non-regenerative heat exchanger, provide the recommended verification of the effectiveness of the Water Chemistry program to mitigate cracking of stainless steel heat exchanger components exposed to treated borated water greater than 60°C (140°F).

LRA Section 3.3.2.2.4.1, Table 3.3.1, Table 3.3.2-5 and Appendix B, Section B.2.1.20 are revised as shown in Enclosure B of this letter.

RAI 3.3.2.2-02

Background:

SRP-LR paragraph 3.3.2.2.4.1 addresses cracking due to stress corrosion cracking and cyclic loading for regenerative heat exchanger components exposed to treated borated water greater than 60°C. The GALL Report recommends the Water Chemistry Program to manage cracking and a plant-specific program to verify the effectiveness of the Water Chemistry Program.

Issue:

The staff noted that the applicant is using the One-Time Inspection Program in lieu of a plant-specific program where periodic inspections are to be scheduled and note A is assigned to the One-Time Inspection for regenerative heat exchangers.

Request:

Provide justification for using one-time inspection in lieu of periodic inspections of a plant-specific program and the use of Note A instead of Note E when applying the One-Time Inspection Program to verify the effectiveness of the Water Chemistry Program.

PSEG Response:

As described in the LRA Section 3.3.2.2.4.2, the Water Chemistry aging management program (Salem LRA Appendix B, Section B.2.1.2) and the One-Time Inspection aging management program (Salem LRA Appendix B, Section B.2.1.20) will be used to manage cracking due to stress corrosion cracking and cyclic loading for the stainless steel regenerative heat exchanger components exposed to treated borated water greater than 60°C (140°F) in the Chemical & Volume Control System.

The regenerative heat exchanger is all-welded construction that prevents heat exchanger disassembly for access to the heat exchanger internals without cutting the piping. This heat exchanger consists of three separate once-through shells connected by welded pipe only and no bolted connections. During refueling outages, the area around the regenerative heat exchanger remains a very high dose area with a background radiation level in excess of five Rem per hour. The One-Time Inspection aging management program will include an inspection of a non-regenerative heat exchanger in the Chemical & Volume Control System with stainless steel components in a similar environment to confirm this aging effect is adequately managed by the Water Chemistry aging management program. The regenerative heat exchanger environment is unique, since both sides of the heat exchanger are treated borated water greater than 60°C (140°F). The non-regenerative heat exchangers have treated borated water greater than 60°C (140°F) inside the tubes and therefore provide an inspection location that is representative for the regenerative heat exchanger components.

The regenerative heat exchanger is an extension of the reactor coolant system in the letdown and charging flow path, which is monitored by the reactor coolant system leakage surveillance. This leakage is normally monitored daily, and the surveillance interval does not exceed 72 hours. The acceptance criterion for the unidentified leakage is one gallon per minute in the Technical Specification surveillance. In addition, the integrity of the regenerative heat exchanger

tubes is verified by tube-side continuous temperature monitoring every shift by the licensed control room operators. The Water Chemistry aging management program and the One-Time Inspection aging management program are discussed in Appendix B. A search of the plant operating experience has found no instances of tube leakage for the regenerative heat exchanger.

Note A was inadvertently used in conjunction with the One-Time Inspection aging management program as the credited GALL-recommended plant specific program. Note A is revised to Note E for the regenerative heat exchanger line items, as shown in Enclosure B to this letter. The One-Time Inspection aging management program provides the means to verify the effectiveness of the Water Chemistry Program for managing the aging effect due to cracking without the excessive dose that the Periodic Inspection aging management program would require. The One-Time Inspection aging management program is the acceptable method of confirming the effectiveness of the Water Chemistry Program in accordance with NUREG-1801, Volume 2, Section XI.M32.

As a result of the correction identified above, the first paragraph of Section 3.3.2.2.4.2 is revised as shown in Enclosure B of this letter.

RAI 3.3.2.12-01

Background:

LRA Table 3.3.2-12, pages 3.3-241 and 242, states that gray cast iron (retarding chamber) tanks exposed to raw water have an aging effect of loss of material due to general, pitting, crevice, galvanic, and microbiologically influenced corrosion, and fouling; and that the aging effects will be managed by the Fire Water System Program. The aging management review (AMR) line items reference Table 1 line item 3.3.1-68 and GALL Report item VII.G-24, which is for piping, piping components, and piping elements, and cite generic note C, indicating that the component is different, but consistent with NUREG-1801 item for material, environment, and aging effect.

Issue:

The staff reviewed GALL AMP XI.M27, Fire Water System and noted that the aging management program (AMP) recommends wall thickness evaluations of fire protection piping be performed using non-intrusive technique (e.g., volumetric testing) to identify loss of material due to corrosion. Based on a telephone conversation with the applicant on May 25, 2010, the applicant clarified that the retarding chamber is a vertical pipe installed in horizontal piping system and acts like a tank. The staff determines that this vertical pipe is a low point in the system and will have stagnant water and will be susceptible to loss of material due to general, pitting and crevice corrosion. It is not clear from a review of LRA Section B.2.1.16, Fire Water System Program, whether volumetric inspection to detect loss of material due to corrosion will be performed on the internal surfaces (specifically the bottom) of the fire water and the retarding chamber tanks.

Request:

Clarify if the (retarding chamber) tanks are included in the sample of fire protection system components that will be volumetrically inspected for wall thickness evaluation to detect loss of material prior to loss of intended function. If not included, please justify how loss of material due to general, pitting, crevice, galvanic, and microbiologically influenced corrosion, and fouling will be detected for these tanks by the Fire Water System Program.

PSEG Response:

Review of Salem component databases and plant drawings determined that retarding chamber tanks are not utilized in portions of the Fire Protection System that are in the scope of license renewal. The function that would be performed by the retarding chamber tanks is achieved in the in-scope portions of the Fire Protection System with an electrical time delay. The retarding chamber tanks from the not in-scope portions of the Fire Protection System were inadvertently included in LRA Table 2.3.3.12 and LRA Table 3.3.2-12. Therefore, retarding chamber tanks are not included in the sample of fire protection system components that will be volumetrically inspected for wall thickness. However, as stated in the Salem LRA, Appendix B, Section B.2.1.16, page B-82, the Fire Water System aging management program will inspect portions of the water based Fire Protection System piping located above ground and exposed to the water internal environment by non-intrusive volumetric examination.

LRA Tables 2.3.3.12 and 3.3.2-12 are revised as shown in Enclosure B of this letter.

RAI 3.4.1-01

Background:

The SRP-LR, Table 3.4-1, item 16, states that stainless steel piping, piping components, piping elements, tanks, and heat exchanger components exposed to treated water are subject to loss of material due to pitting and crevice corrosion. The GALL Report, under item VIII.B1-4, recommends managing the aging effect using the Water Chemistry and One-Time Inspection Programs.

Issue:

The Salem Nuclear Generating Station LRA Table 3.1.2-4 indicates that stainless steel steam generator tube support plates exposed to treated water greater than 140 °F can undergo loss of material due to pitting and crevice corrosion, and the Steam Generator Tube Integrity and Water Chemistry Programs will be used to manage this aging effect. The AMR line items cite generic note E, indicating that they are consistent with the GALL Report item for material, environment and aging effect, but a different aging management program is credited. The LRA references Table 3.4.1, item 3.4.1-16 and GALL Report item VIII.B1-4 for consistency with the GALL Report. However, GALL Report item VIII.B1-4 recommends managing the aging effect using the Water Chemistry Program augmented by the One-Time Inspection Program to verify the effectiveness of the chemistry control program.

Request:

Clarify how the AMR line items in LRA Table 3.1.2-4 for tube support plates loss of material due to pitting and crevice corrosion are consistent with the GALL Report item VIII.B1-4 for piping, piping components, and piping elements and provide a basis for the adequacy of the Steam Generator Tube Integrity Program to verify the effectiveness of the Water Chemistry Program for these components.

PSEG Response:

Salem selected GALL Report item VIII.B1-4 (piping, piping components, and piping elements) since the stainless steel component type Steam Generators (Tube Support Plates) was subjected to the treated water > 140°F environment, and therefore considered this material, environment, and aging effects combination most closely aligned to this item number. Salem had substituted the Steam Generator Tube Integrity aging management program (Salem LRA Appendix B, Section B.2.1.10) for the One-Time Inspection aging management program (Salem LRA Appendix B, Section B.2.1.20) in Table 3.1.2-4, "Steam Generators", since this aging management program inspects the Steam Generator tube support plates during refueling outages. However, to be consistent with the GALL Report item VIII.B1-4, Salem will use the One-Time Inspection aging management program for inspections of the Steam Generator tube support plates to verify the effectiveness of the Salem Water Chemistry aging management program (Salem LRA Appendix B, Section B.2.1.2).

LRA Table 3.1.2-4, Page 3.1-143, "Steam Generators", is revised as shown in Enclosure B of this letter.

RAI 3.5.2.2.1.7-01

Background:

The GALL Report, under items II.A3-2 and II.B4-2, indicates that stress corrosion cracking can occur for stainless steel penetration sleeves, penetration bellows, and associated welds exposed to air – indoor uncontrolled or air - outdoor. For stress corrosion cracking to occur, a susceptible material must be subjected to a tensile stress and a corrosive environment. The GALL Report indicates that these three conditions can exist and cause stress corrosion cracking for stainless steel penetration sleeves, bellows and welds.

Issue:

In terms of the aging management of stainless steel penetration sleeves, penetration bellows and associated welds, LRA Table 3.5.1, item 3.5.1-10 states "stress corrosion cracking will not occur at these components, within the scope of the license renewal, because the normal stress and environmental exposure conditions are not conducive to the development of stress corrosion cracking."

However, LRA Table 3.5.2-3 on page 3.5-179 addresses loss of material due to pitting and crevice corrosion in the stainless steel penetration sleeves (cap plates) exposed to air with steam or water leakage. The applicant credited the 10 CFR Part 50, Appendix J Program and the ASME Section XI, Subsection IWE Program to manage loss of material. LRA Note 3 on page 3.5-187 states "air with steam or water leakage environment is applicable to local areas inside the containment that are exposed to potential service water leakage or spray." LRA Note 3 on page 3.5-187 also states "plant operating experience showed that metal components in this environment exhibit aging effects observed in Air-Outdoor environment." Therefore, the aging management review results of the applicant are in potential conflict with the applicant's claim that the normal environmental conditions are not conducive to the development of stress corrosion cracking.

LRA Section 3.5.2.2.1.7 further states "the containment pressure boundary welds between stainless steel piping and penetration sleeves, with normal operating temperatures above 140 degrees F, are not highly stressed." However, the LRA does not provide a detailed technical basis for the applicant's claim that the penetration sleeves are not highly stressed so that the normal stress conditions are not conducive to the development of stress corrosion cracking. The LRA does not provide the information on how the applicant evaluated the residual stresses in the welds and adjacent material.

Request:

1. Describe detailed operating experience in terms of the observation of pitting and crevice corrosion and stress corrosion cracking in the penetrations (penetration sleeves, bellows and welds) and determine whether the operating experience supports the applicant's claim that the normal stress and environmental exposure conditions are not conducive to the development of stress corrosion cracking in these components.

2. Clarify why the applicant claims that the environmental condition is not conducive to the development of stress corrosion cracking in the containment penetrations although (1) the penetration sleeves are exposed to air with steam or water leakage due to potential service water leakage or spray, (2) the applicant manages loss of material due to pitting and crevice corrosion in the penetration sleeves exposed to the environment, and (3) plant operating experience showed that metal components in this environment exhibit aging effects observed in Air-Outdoor environment (LRA Page 3.5-187, Note 3).

3. Describe how the applicant determined that the welds between the stainless steel piping and penetration sleeves are not highly stressed. In addition, clarify whether the stress evaluation includes residual stresses and clarify whether the condition including residual stresses is not conducive to the development of stress corrosion cracking in the containment penetrations.

4. Based on the information provided for Requests 1 through 3, justify why stress corrosion cracking is not applicable to the stainless steel penetrations (penetration sleeves, penetration bellows and associated welds) that are addressed in LRA Table 3.5.1, item 3.5.1-10.

PSEG Response:

1. Salem Unit 1 and Unit 2 containment piping penetration configurations are as shown in UFSAR Figures 3.8-49 and 3.8-50. As indicated in Figure 3.8-49, hot pipe penetrations were installed with stainless steel expansion bellows on the outside of the containment and the bellows are not a part of the containment pressure boundary. The cold pipe penetrations, Figure 3.8-50, have no expansion bellows. In both cases, the containment pressure boundary is formed by the penetration sleeve, cap plates, piping and the associated welds inside the containment. The pipe penetration sleeves are constructed of carbon steel, while the cap plates are stainless steel or carbon steel. The containment penetration pressure boundary welds of interest are the cap plate to penetration sleeve for those cap plates constructed of stainless steel (dissimilar metal welds), and the cap plate to penetrating pipe, for penetrating pipe constructed of stainless steel. Penetration sleeves and the cap plates inside the containment are exposed to normal operating PWR containment atmosphere environment. The "Air with Steam or Water Leakage" environment was conservatively assumed for penetration sleeves and cap plates where leakage has occurred from brackish water systems.

   A search of all available data in the corrective action database was performed for operating experience related to the penetration pressure boundary welds of interest. The search criteria included all conditions reported on penetrations capturing any findings as a result of ASME Section XI, Subsection IWE inspections as well as any other source. There were no items associated with pitting, crevice corrosion, or stress corrosion cracking of the stainless steel containment penetration pressure boundary components or welds.

   Liquid Penetrant (PT) surface examinations (as required by Inservice Inspection for IWB and IWC integral attachments) have been performed on nine of the cap plate to penetrating pipe welds for stainless steel penetrating pipe and no flaws were found in these welds. The inspections performed include 11% of the population of those welds with a normal operating temperature of greater than 140 degrees F.

Nine Appendix J Type "A" Containment Integrated Leak Rate Tests, five for Unit 1 and four for Unit 2, have been performed with acceptable results, and did not reveal any degradation of these containment penetration pressure boundary welds.

In summary, the operating experience reviews, Liquid Penetrant (PT) surface examinations, and the Appendix J Type "A" tests associated with the stainless steel and dissimilar metal containment penetration pressure boundary welds have not revealed any indications of pitting, crevice corrosion, or stress corrosion cracking. This operating experience supports the position that normal stress and environmental exposure conditions are not conducive to the development of stress corrosion cracking in these components. However, the LRA is revised to add cracking due to stress corrosion cracking for the stainless steel and dissimilar metal containment penetration pressure boundary welds, as shown in Enclosure B.

2. Operating experience reviews revealed that both Salem Units 1 and 2 have previously experienced Containment Fan Cooling Unit (CFCU) leaks in the containment. This is consistent with the environment and aging effect stated in Salem LRA Table 3.5.2-3 Containment Structure for the penetration sleeves, "Air with Steam or Water Leakage", and "loss of material due to pitting and crevice corrosion" and the associated Plant Specific Note 3 which states:

   "Air with steam or water leakage environment is applicable to local areas inside the containment that are exposed to potential service water leakage or spray. Plant operating experience showed that metal components in this environment exhibit aging effects observed in Air-Outdoor environment."

   These previous event driven leaks have introduced a potential to have exposed the containment penetration pressure boundary stainless steel and dissimilar metal welds to brackish water leakage. This potential existed for those containment penetrations located under the CFCUs. Specifically, there have been occurrences where Service Water has leaked from the upper elevations of the containment building where the CFCUs and associated motor cooler units are located. The elevations in the containment are separated for the most part by solid floors but there are gaps between the floors and the containment wall that introduces a potential pathway to the containment penetrations located on the lower elevations. Corrective actions taken as a result of the past leaks included modifications which were implemented in 2008 at both units that substantially reduced the probability of water hammer events, which were the cause of the majority of the previous leaks. In addition, as a result of the events and site sensitivity to the potential adverse affects of the service water leakage, a comprehensive event driven service water spill response procedure was developed to mitigate any possible similar events in the future. The procedure includes requirements for walk downs, swipe sampling for chlorides, flushing, cleaning with demineralized water and re-swiping to assure residual chlorides are below an acceptable level. Since there was a potential for an adverse environment at the penetrations in the past, it cannot be concluded that the environmental conditions conducive to the development of stress corrosion cracking were never present. However, the operating experience reviews, Liquid Penetrant (PT) surface examinations, and the Appendix J Type "A" tests associated with the stainless steel and dissimilar metal containment penetration pressure boundary welds have not revealed any indications of stress corrosion cracking.

3. Calculations that verified the adequacy of the design for the piping penetrations that penetrate the reactor containment wall for Salem Units 1 and 2 indicate the stresses on the piping penetrations and associated welds are low when compared to the allowable stresses. Residual weld stresses were not included in the evaluation that concluded the piping and penetration sleeves are not highly stressed. Since the threshold for the minimum level of tensile stress necessary for stress corrosion cracking to occur cannot be quantified, and is also dependent on the relative severity of the corrosion-affecting parameters and other factors, it cannot be concluded with absolute certainty that the residual stresses present are not conducive to the development of stress corrosion cracking for the stainless steel and dissimilar metal welds associated with the containment penetration pressure boundary welds.

4. Based upon the response to item 1 above, the components of interest are the welds associated with the cap plate to penetration sleeve for those cap plates constructed of stainless steel, and the cap plate to penetrating pipe, for penetrating pipe constructed of stainless steel. Operating experience reviews with searches that would capture items reported as a result of ASME Section XI, Subsection IWE inspections as well as any other source, Liquid Penetrant (PT) surface examinations of 11% of the welds of interest, and the Appendix J Type "A" tests associated with the stainless steel and dissimilar metal containment penetration pressure boundary welds have not revealed any indications of stress corrosion cracking. However, based on the information provided above, it has been concluded there is a potential for stress corrosion cracking of the stainless steel and dissimilar metal welds associated with the containment penetration cap plate pressure boundary welds that are subjected to normal operating temperatures greater than 140 degrees F.

   The aging effect and mechanism of cracking due to stress corrosion cracking is considered potentially applicable for the penetration sleeve (cap plate) components that involve stainless steel material with normal operating temperatures greater than 140 degrees F, exposed to an air with steam or water leakage environment in the Containment Structure.

As a result of this response, Salem LRA Section 3.5.2.1.3, Section 3.5.2.2.1.7, Table 3.5.1, Table 3.5.2-3, Appendix A Sections A.2.1.28 and A.2.1.31, and Appendix B Sections B.2.1.28 and B.2.1.31 are revised as shown in Enclosure B.

Enclosure B

Salem Generating Station Units 1 and 2 License Renewal Application Updates

Note: To facilitate understanding, portions of the original LRA have been repeated in this Enclosure, with revisions indicated. Existing LRA text is shown in normal font. Changes are highlighted with bolded italics for inserted text and strikethroughs for deleted text.

As a result of the response to **RAI 3.3.2.2-01** provided in Enclosure A of this letter, Section 3.3.2.2.4.1, Table 3.3.1, Table 3.3.2-5 and Appendix B, Section B.2.1.20 are revised as shown below.

3.3.2.2.4 Cracking due to Stress Corrosion Cracking and Cyclic Loading

1. *Cracking due to SCC and cyclic loading could occur in stainless steel PWR non-regenerative heat exchanger components exposed to treated borated water greater than 60°C (>140°F) in the chemical and volume control system. The existing aging management program on monitoring and control of primary water chemistry in PWRs to manage the aging effects of cracking due to SCC. However, control of water chemistry does not preclude cracking due to SCC and cyclic loading. Therefore, the effectiveness of the water chemistry control program should be verified to ensure that cracking is not occurring. The GALL Report recommends that a plant-specific aging management program be evaluated to verify the absence of cracking due to SCC and cyclic loading to ensure that these aging effects are managed adequately. An acceptable verification program is to include temperature and radioactivity monitoring of the shell side water, and eddy current testing of tubes.*

   Salem will implement the Water Chemistry program, B.2.1.2. ~~while~~ ***Effectiveness of the Water Chemistry program is verified by periodically monitoring reactor coolant leakage for the tube side and*** continuously monitoring for radioactivity on the shell side of the non-regenerative stainless steel heat exchangers***,*** ~~to manage the aging effects. Shutdown eddy current inspection activities would result in significant radiation exposure to plant personnel.~~ ***and by a One-Time Inspection aging management program activity to perform eddy current testing of stainless steel tubes in a non-regenerative heat exchanger exposed to treated borated water greater than 60°C (140°F).***

   The non-regenerative heat exchanger radiation instrumentation is continuously monitored and tube leakage would be detected by abnormal radiation indications ***on the Component Cooling System. Also, the reactor coolant system leakage is normally measured daily and at least once per 72 hours with a plant shutdown limit of 1.0 gpm unidentified leakage.*** The Water Chemistry program activities provide for monitoring and controlling the chemical environments of the primary cycle systems in accordance with EPRI, Pressurized Water Reactor Primary Chemistry Guidelines. The Water Chemistry program activities mitigate loss of material, reduction of heat transfer and cracking aging effects to ensure there is no loss of component intended function. The Water Chemistry aging management program ***and One-Time Inspection aging management program are*** described in Appendix B.

Table 3.3.1 Summary of Aging Management Evaluations for the Auxiliary Systems

Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.3.1-7	Stainless steel non-regenerative heat exchanger components exposed to treated borated water >60°C (>140°F)	Cracking due to stress corrosion cracking and cyclic loading	Water Chemistry and a plant-specific verification program. An acceptable verification program is to include temperature and radioactivity monitoring of the shell side water, and eddy current testing of tubes.	Yes, plant specific	~~Consistent with NUREG-1801.~~ The Water Chemistry program and ***the One-Time Inspection program*** will be used to manage cracking due~~s~~ to stress corrosion cracking and cyclic loading of the stainless steel heat exchanger components exposed to treated borated water ***greater than*** 60°C (140°F) in the Component Cooling System. Continuous radioactivity monitoring on the shell side of the heat exchangers ***with periodic reactor coolant leakage monitoring on the tube side, and a One-Time Inspection program activity to perform eddy current testing of stainless steel tubes in a non-regenerative heat exchanger exposed to treated borated water greater than 60°C (140°F),*** provides verification of the Water Chemistry program effectiveness in managing the aging effects for these components. See subsection 3.3.2.2.4.1.

Table 3.3.2-5 Component Cooling System

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
Heat Exchanger Components (Excess Letdown)	*Pressure Boundary*	*Stainless Steel (Tubes)*	*Treated Borated Water (Internal) > 140 F*	*Cracking/Stress Corrosion Cracking, Cyclic Loading*	*One-Time Inspection*	*VII.E1-9*	*3.3.1-7*	*E, 8*
Heat Exchanger Components (Excess Letdown)	*Pressure Boundary*	*Stainless Steel (Tubesheet)*	*Treated Borated Water (Internal) > 140 F*	*Cracking/Stress Corrosion Cracking, Cyclic Loading*	*One-Time Inspection*	*VII.E1-9*	*3.3.1-7*	*E, 8*
Heat Exchanger Components (Excess Letdown)	*Pressure Boundary*	*Stainless Steel (Tubeside Components)*	*Treated Borated Water (Internal) > 140 F*	*Cracking/Stress Corrosion Cracking, Cyclic Loading*	*One-Time Inspection*	*VII.E1-9*	*3.3.1-7*	*E, 8*
Heat Exchanger Components (Letdown)	*Pressure Boundary*	*Stainless Steel (Tubes)*	*Treated Borated Water (Internal) > 140 F*	*Cracking/Stress Corrosion Cracking, Cyclic Loading*	*One-Time Inspection*	*VII.E1-9*	*3.3.1-7*	*E, 8*
Heat Exchanger Components (Letdown)	*Pressure Boundary*	*Stainless Steel (Tubesheet)*	*Treated Borated Water (Internal) > 140 F*	*Cracking/Stress Corrosion Cracking, Cyclic Loading*	*One-Time Inspection*	*VII.E1-9*	*3.3.1-7*	*E, 8*
Heat Exchanger Components (Letdown)	*Pressure Boundary*	*Stainless Steel (Tubeside)*	*Treated Borated Water (Internal) > 140 F*	*Cracking/Stress Corrosion Cracking, Cyclic Loading*	*One-Time Inspection*	*VII.E1-9*	*3.3.1-7*	*E, 8*
Heat Exchanger Components (Seal Water)	*Pressure Boundary*	*Stainless Steel (Tubes)*	*Treated Borated Water (Internal) > 140 F*	*Cracking/Stress Corrosion Cracking, Cyclic Loading*	*One-Time Inspection*	*VII.E1-9*	*3.3.1-7*	*E,8*
Heat Exchanger Components (Seal Water)	*Pressure Boundary*	*Stainless Steel (Tubesheet)*	*Treated Borated Water (Internal) > 140 F*	*Cracking/Stress Corrosion Cracking, Cyclic Loading*	*One-Time Inspection*	*VII.E1-9*	*3.3.1-7*	*E, 8*
Heat Exchanger Components (Seal Water)	*Pressure Boundary*	*Stainless Steel (Tubeside Components)*	*Treated Borated Water (Internal) > 140 F*	*Cracking/Stress Corrosion Cracking, Cyclic Loading*	*One-Time Inspection*	*VII.E1-9*	*3.3.1-7*	*E, 8*

Plant Specific Notes:

8. NUREG-1801 specifies a plant-specific program. The One-Time Inspection program is used to manage the aging effect(s) applicable to this component type, material, and environment combination.

B.2.1.20 One-Time Inspection

Program Description

The inspection sample includes one-time inspection of more susceptible materials in potentially more aggressive environments (e.g., low or stagnant flow areas) to manage the effects of aging. Qualified personnel following station procedures that are based on applicable codes and standards, including ASME, and 10 CFR 50, Appendix B, will perform the inspections. Examination methods will include volumetric examination ~~(UT or RT)~~ or visual examination (VT-1, VT-3, or equivalent). Acceptance criteria are in accordance with industry guidelines, codes, and standards, including ASME Boiler and Pressure Vessel Code, Section XI, 1998 Edition, 2000 Addenda.

As a result of the response to **RAI 3.3.2.2-02** provided in Enclosure A of this letter, LRA Section 3.3.2.2.4.2, Table 3.3.1 and Table 3.3.2-2 are revised as shown below.

3.3.2.2.4 Cracking due to Stress Corrosion Cracking and Cyclic Loading

2. *Cracking due to SCC and cyclic loading could occur in stainless steel PWR regenerative heat exchanger components exposed to treated borated water greater than 60°C (>140°F). The existing aging management program relies on monitoring and control of primary water chemistry in PWRs to manage the aging effects of cracking due to SCC. However, control of water chemistry does not preclude cracking due to SCC and cyclic loading. Therefore, the effectiveness of the water chemistry control program should be verified to ensure that cracking is not occurring. The GALL Report recommends that a plant-specific aging management program be evaluated to verify the absence of cracking due to SCC and cyclic loading to ensure that these aging effects are managed adequately. Acceptance criteria are described in Branch Technical Position RLSB-1.*

   Salem will implement the Water Chemistry program, B. 2.1.2, ***and One-Time Inspection program, B.2.1.20,*** to manage cracking due to stress corrosion cracking and cyclic loading for the stainless steel regenerative heat exchanger components exposed to treated borated water greater than 60°C (140°F) in the Chemical & Volume Control system. Integrity of the regenerative heat exchanger is verified by ***periodic reactor coolant system leakage monitoring and*** continuous temperature monitoring. ***The regenerative heat exchanger is all-welded construction and inspections are not possible without significant dose.*** The Water Chemistry program activities provide for monitoring and controlling the chemical environments of the primary cycle systems in accordance with EPRI, Pressurized Water Reactor Primary Chemistry Guidelines. The Water Chemistry program activities mitigate cracking to ensure there is no loss of component intended function. The One-Time Inspection program includes inspections of other stainless steel components in this environment to verify the effectiveness of the Water Chemistry program to manage cracking and other aging effects. ***Specifically, one of the non-regenerative heat exchangers will be inspected to confirm the condition of the tubes.*** The One-Time Inspection aging management program and Water Chemistry aging management program are described in Appendix B.

Table 3.3.1 Summary of Aging Management Evaluations for the Auxiliary Systems

Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.3.1-8	Stainless steel regenerative heat exchanger components exposed to treated borated water >60°C (>140°F)	Cracking due to stress corrosion cracking and cyclic loading	Water Chemistry and a plant-specific verification program. The AMP is to be augmented by verifying the absence of cracking due to stress corrosion cracking and cyclic loading. A plant specific aging management program is to be evaluated.	Yes, plant specific	~~Consistent with NUREG-1801~~. The Water Chemistry program, B.2.1.2, ***and the One-Time Inspection program, B.2.1.20,*** will be used to manage cracking due to stress corrosion cracking and cyclic loading of the stainless steel regenerative heat exchanger components exposed to treated borated water ***greater than*** 60°C (140°F) in the Chemical and Volume Control system. The Water Chemistry program effectiveness is verified by one-time inspection of other stainless steel components in this environment. Integrity of the regenerative heat exchanger is verified by ***periodic reactor coolant system leakage and*** continuous temperature monitoring. See subsection 3.3.2.2.4.2.

Table 3.3.2-2 Chemical & Volume Control System (Continued)

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
Heat Exchanger Components (Regenerative)	Pressure Boundary	Stainless Steel (Shellside Components)	Treated Borated Water (Internal) > 140 F	Cracking/Stress Corrosion Cracking, Cyclic Loading	One-Time Inspection	VII.E1-5	3.3.1-8	~~A~~ ***E,8***
Heat Exchanger Components (Regenerative)	Pressure Boundary	Stainless Steel (Tubes)	Treated Borated Water (External) > 140 F	Cracking/Stress Corrosion Cracking, Cyclic Loading	One-Time Inspection	VII.E1-5	3.3.1-8	~~A~~ ***E,8***
Heat Exchanger Components (Regenerative)	Pressure Boundary	Stainless Steel (Tubes)	Treated Borated Water (Internal) > 140 F	Cracking/Stress Corrosion Cracking, Cyclic Loading	One-Time Inspection	VII.E1-5	3.3.1-8	~~A~~ ***E,8***
Heat Exchanger Components (Regenerative)	Pressure Boundary	Stainless Steel (Tubesheet)	Treated Borated Water (External) > 140 F	Cracking/Stress Corrosion Cracking, Cyclic Loading	One-Time Inspection	VII.E1-5	3.3.1-8	~~A~~ ***E,8***
Heat Exchanger Components (Regenerative)	Pressure Boundary	Stainless Steel (Tubesheet)	Treated Borated Water (Internal) > 140 F	Cracking/Stress Corrosion Cracking, Cyclic Loading	One-Time Inspection	VII.E1-5	3.3.1-8	~~A~~ ***E,8***
Heat Exchanger Components (Regenerative)	Pressure Boundary	Stainless Steel (Tubeside Components)	Treated Borated Water (Internal) > 140 F	Cracking/Stress Corrosion Cracking, Cyclic Loading	One-Time Inspection	VII.E1-5	3.3.1-8	~~A~~ ***E,8***

Plant Specific Notes:

8. NUREG-1801 specifies a plant specific program. The One-Time Inspection aging management program is used to manage the aging effect(s) applicable to this component type, material, and environment combination.

As a result of the response to **RAI 3.3.2.12-01** provided in Enclosure A of this letter, LRA Table 2.3.2-12 and Table 3.3.2-12 are revised as shown below.

Table 2.3.3-12 Fire Protection System
Components Subject to Aging Management Review

Component Type	Intended Function
~~Tanks (Retarding Chamber)~~	~~Pressure Boundary~~

Table 3.3.2-12 Fire Protection System

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-1801 Vol.2 Item	Table 1 Item	Notes
~~Tanks (Retarding Chamber)~~	~~Pressure Boundary~~	~~Carbon Steel~~	~~Air – Indoor (External)~~	~~Loss of Material/ General Corrosion~~	~~External Surfaces Monitoring~~	~~VII.I-8~~	~~3.3.1-58~~	~~A~~
~~Tanks (Retarding Chamber)~~	~~Pressure Boundary~~	~~Carbon Steel~~	~~Air – Indoor (External)~~	~~Loss of Material/ General Corrosion~~	~~Fire Water System~~	~~VII.I-8~~	~~3.3.1-58~~	~~E, 8~~
~~Tanks (Retarding Chamber)~~	~~Pressure Boundary~~	~~Carbon Steel~~	~~Raw Water (Internal)~~	~~Loss of Material/ General, Pitting, Crevice, and Microbiologically Influenced Corrosion, and Fouling~~	~~Fire Water System~~	~~VII.G-24~~	~~3.3.1-68~~	~~C~~

As a result of the response to **RAI 3.4.1-01** provided in Enclosure A of this letter, LRA Table 3.1.2-4 is revised as shown below.

Table 3.1.2-4 Steam Generators

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-1801 Vol. 2 Item	Table 1 Item
Steam Generators (Tube Support Plates)	Structural Support	Stainless Steel	Treated Water (External) > 140 F	Loss of Material/Pitting and Crevice Corrosion	~~Steam Generator Tube Integrity~~ ***One-Time Inspection***	***VIII.B1-4***	***3.4.1-16***

As a result of the response to **RAI 3.5.2.2.1.7-01** provided in Enclosure A of this letter, Salem LRA Section 3.5.2.1.3, Section 3.5.2.2.1.7, Table 3.5.1, Table 3.5.2-3, Appendix A Sections A.2.1.28 and A.2.1.31, and Appendix B Sections B.2.1.28 and B.2.1.31 are revised as shown below.

3.5.2.1.3 Containment Structure

Aging Effects Requiring Management

The following aging effects associated with the Containment Structure components require management:

- ***Cracking/Stress Corrosion Cracking***

3.5.2.2.1.7 Cracking due to Stress Corrosion Cracking (SCC)

Cracking due to stress corrosion cracking of stainless steel penetration sleeves, penetration bellows, and dissimilar metal welds could occur in all types of PWR and BWR containments. Cracking due to SCC could also occur in stainless steel vent line bellows for BWR containments. The existing program relies on ASME Section XI, Subsection IWE and 10 CFR Part 50, Appendix J to manage this aging effect. The GALL Report recommends further evaluation of additional appropriate examinations/evaluations implemented to detect these aging effects for stainless steel penetration sleeves, penetration bellows and dissimilar metal welds, and stainless steel vent line bellows.

Item Number 3.5.1-11 is applicable to BWRs only and is not used for Salem.

The following discussion is applicable to Item Number 3.5.1-10.

At Salem, ASME Section XI, Subsection IWE, B.2.1.28, and 10 CFR Part 50 Appendix J, B.2.1.31 programs will be used to manage cracking due to stress corrosion cracking for the stainless steel and dissimilar metal welds associated with the containment penetration pressure boundary with normal operating temperatures greater than 140 degrees F. These welds are the cap plate to penetration sleeve for those cap plates constructed of stainless steel (dissimilar metal welds), and the cap plate to penetrating pipe, for penetrating pipe constructed of stainless steel. Plant operating experience, Liquid Penetrant (PT) surface examinations on 11% of the applicable cap plate to stainless steel penetrating pipe welds and Appendix J Type "A" tests have not revealed any indications of cracking or flaws, therefore, augmented or additional inspections are not warranted.

Stress corrosion cracking (SCC) is not an applicable aging mechanism for the Containment Structure carbon steel penetration sleeves~~,~~ and the stainless steel penetration bellows~~, and dissimilar metal welds~~. The Salem Containment liner and associated penetration sleeves are carbon steel. High temperature piping systems penetrating the containment are generally carbon steel. Stress corrosion cracking is

only applicable to stainless steel. ~~and is predicted only under certain conditions. There are stainless steel and dissimilar metal welds associated with stainless steel piping welded to penetration sleeve cap plates. Salem aging management reviews concluded SCC, of stainless steel at the penetration sleeves, is not considered credible because stainless steel SCC requires a concentration of chloride or sulfate contaminants, which are not normally present in significant quantities, as well as, high stress, and temperatures greater than 140 degrees F. Leakage of water in the containment, which might contact the penetration sleeves, is event driven. The containment pressure boundary welds between stainless steel piping and penetration sleeves, with normal operating temperatures above 140 degrees F, are not highly stressed. In addition, the Technical Specification limits the average air temperature inside the Containment during normal plant operation to 120 degrees F.~~ ***The containment penetration bellows are located outside the containment, where the normal operating temperature is less than 140 degrees F.***

~~Therefore cracking of these components due to stress corrosion cracking is not applicable.~~
Cracking of the Containment stainless steel penetration bellows, due to stress corrosion cracking is ~~also~~ not applicable because the containment penetration bellows are not part of the containment leakage limiting boundary per UFSAR section, 3.8.1.6.8.10, Piping Penetrations:

"....Containment piping penetrations designed for Salem are not required to be type "B" tested for 10CFR50 Appendix J (Ref. Safety Evaluation S-C-R700-MSE-0253 Rev. 0). The type "B" test is applicable to piping penetrations that utilize expansion bellows as the leakage limiting boundary. The piping penetrations at Salem rely on partial/full penetration seal welds inside containment as the leakage limiting boundary, which are leak rate tested as part of the Appendix J type "A" Containment Integrated Leak Rate Test (ILRT)..."

The stainless steel transfer tube bellows are located in the reactor cavity pool and are not subject to the combination of high temperature, high stress and the concentration of contaminants required for stress corrosion cracking. The transfer tube bellows are evaluated for cumulative fatigue as addressed in paragraph 3.5.2.2.1.6 above. A TLAA evaluation for the Transfer Tube bellows was performed. The TLAA is evaluated in accordance with 10 CFR 54.21(c). Evaluation of this TLAA is discussed in Section 4.5 "Fuel Transfer Tube Bellows Design Cycles".

Table 3.5.1 Summary of Aging Management Evaluations for Structures and Component Supports					
Item Number	**Component**	**Aging Effect/ Mechanism**	**Aging Management Programs**	**Further Evaluation Recommended**	**Discussion**
3.5.1-10	Stainless steel penetration sleeves, penetration bellows, dissimilar metal welds	Cracking due to stress corrosion cracking	ISI (IWE) and 10 CFR Part 50, Appendix J, and additional appropriate examinations/ evaluations for bellows assemblies and dissimilar metal welds.	Yes, detection of aging effects is to be evaluated	***ASME Section XI, Subsection IWE, B.2.1.28, and 10 CFR Part 50 Appendix J, B.2.1.31 programs will be used to manage cracking due to stress corrosion cracking for the stainless steel and dissimilar metal welds associated with the containment penetration pressure boundary with normal operating temperatures greater than 140 degrees F.*** ~~Not Applicable. Stress corrosion cracking will not occur at these components, within the scope of license renewal, because the normal stress and environmental exposure conditions are not conducive to the development of stress corrosion cracking.~~ See Subsection 3.5.2.2.1.7.

Table 3.5.2-3 Containment Structure

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
Penetration sleeves (cap plates)	*Pressure Boundary*	*Carbon Steel*	*Air with Steam or Water Leakage*	*Cracking/Stress Corrosion Cracking*	*10 CFR Part 50, Appendix J*			*G, 3, 16*
Penetration sleeves (cap plates)	*Pressure Boundary*	*Carbon Steel*	*Air with Steam or Water Leakage*	*Cracking/Stress Corrosion Cracking*	*ASME Section XI, Subsection IWE*			*G, 3, 16*
Penetration sleeves (cap plates)	*Pressure Boundary*	*Stainless Steel*	*Air with Steam or Water Leakage*	*Cracking/Stress Corrosion Cracking*	*10 CFR Part 50, Appendix J*			*G, 3, 16*
Penetration sleeves (cap plates)	*Pressure Boundary*	*Stainless Steel*	*Air with Steam or Water Leakage*	*Cracking/Stress Corrosion Cracking*	*ASME Section XI, Subsection IWE*			*G, 3, 16*
Penetration sleeves (cap plates)	*Structural Support*	*Carbon Steel*	*Air with Steam or Water Leakage*	*Cracking/Stress Corrosion Cracking*	*10 CFR Part 50, Appendix J*			*G, 3, 16*
Penetration sleeves (cap plates)	*Structural Support*	*Carbon Steel*	*Air with Steam or Water Leakage*	*Cracking/Stress Corrosion Cracking*	*ASME Section XI, Subsection IWE*			*G, 3, 16*
Penetration sleeves (cap plates)	*Structural Support*	*Stainless Steel*	*Air with Steam or Water Leakage*	*Cracking/Stress Corrosion Cracking*	*10 CFR Part 50, Appendix J*			*G, 3, 16*
Penetration sleeves (cap plates)	*Structural Support*	*Stainless Steel*	*Air with Steam or Water Leakage*	*Cracking/Stress Corrosion Cracking*	*ASME Section XI, Subsection IWE*			*G, 3, 16*

Plant Specific Notes:

16. ASME Section XI, Subsection IWE, B.2.1.28, and 10 CFR Part 50 Appendix J, B.2.1.31, aging management programs will be used to manage cracking due to stress corrosion cracking for the stainless steel and dissimilar metal welds associated with the containment penetration pressure boundary with normal operating temperatures greater than 140 degrees F. These welds are the cap plate to penetration sleeve for those cap plates constructed of stainless steel (dissimilar metal welds), and the cap plate (carbon or stainless steel) to penetrating pipe, for penetrating pipe constructed of stainless steel. These line items are discussed further in section 3.5.2.2.1.7 of the LRA.

A.2.1.28 ASME Section XI, Subsection IWE

The ASME Section XI, Subsection IWE aging management program is an existing program based on ASME Code and complies with the provisions of 10 CFR 50.55a. The program consists of periodic inspection of the containment structure liner plate, including its integral attachments, penetration sleeves, pressure retaining bolting, personnel airlock and equipment hatches, moisture barrier, and other pressure retaining components for ***cracking***, loss of material, loss of preload, and loss of sealing (of the moisture barrier). The moisture barrier is a sealant installed at the junction of the Containment concrete floor and the carbon steel Containment liner.

The ASME Section XI, Subsection IWE aging management program will be enhanced to include:

3. ***ASME Section XI, Subsection IWE program scope will be revised to include the following welds that are currently exempted from Subsection IWE and governed under ASME Section XI, Subsection IWB or IWC. The scope of the revision will include the cap plate to penetrating pipe pressure boundary welds, for penetrating pipe constructed of stainless steel for those penetrations with a normal operating temperature greater than 140 degrees F.***

A.2.1.31 10 CFR 50, Appendix J

The 10 CFR 50, Appendix J aging management program is an existing program that monitors leakage rates through the containment pressure boundary, including penetrations, fittings and other access openings, in order to detect age related degradation of the containment pressure boundary. Corrective actions are taken if leakage rates exceed acceptance criteria. The Primary Containment Leakage Rate Testing Program (LRT) provides for aging management of pressure boundary degradation due to aging effects from ~~the~~ ***cracking***, loss of leakage tightness, loss of sealing, loss of material, or loss of preload in various systems penetrating containment. The 10 CFR 50 Appendix J program also detects age related degradation in material properties of gaskets, o-rings and packing materials for the containment pressure boundary access points. Consistent with the current licensing basis, the containment leakage rate tests are performed in accordance with the regulations and guidance provided in 10 CFR 50 Appendix J Option B, Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program," NEI 94-01 "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50 Appendix J", and ANSI/ANS 56.8, "Containment System Leakage Testing Requirements."

B.2.1.28 ASME Section XI, Subsection IWE

Program Description

The program is implemented through procedures that implement ASME Section XI, Subsection IWE requirements for detecting ***cracking***, loss of material, loss of preload, and loss of sealing (of the moisture barrier). The containment environments addressed are: air-indoor, air with borated water leakage and air with steam or water leakage.

The program utilizes inspections that detect degradation before loss of intended function. No preventive attributes are associated with these activities. The program implements the requirements of IWE by providing general visual examinations, detailed visual examinations and augmented inspections as approved by 10 CFR 50.55a(a)(3)(i) for evidence of aging effects that could affect structural integrity or leak tightness of the containment structure. Areas subject to augmented inspection are subject to detailed visual inspection and volumetric (ultrasonic) examination techniques as required by engineering. The program addresses the E-A and E-C Examination Categories described in Table IWE-2500-1 in the ASME Code 1998 Edition with 1998 Addenda, as approved per 10 CFR 50.55a. The ASME 1998 and later ASME Codes do not contain categories E-B, E-D, E-F, E-G and E-P. The program specifies examinations of accessible surfaces to detect the aging effects of ***cracking***, loss of material, loss of preload, and loss of sealing (of the moisture barrier) consistent with IWE-3500. The frequency and scope of examinations specified is in accordance with 10 CFR 50.55a and ASME Section XI, Subsection IWE-2400.

Enhancements

Prior to the period of extended operation, the following enhancements will be implemented:

3. ***ASME Section XI, Subsection IWE program scope will be revised to include the following welds that are currently exempted from Subsection IWE and governed under ASME Section XI, Subsection IWB or IWC. The scope of the revision will include the cap plate to penetrating pipe pressure boundary welds, for penetrating pipe constructed of stainless steel for those penetrations with a normal operating temperature greater than 140 degrees F.***

B.2.1.31 10 CFR 50, Appendix J

Program Description

The 10 CFR Part 50, Appendix J aging management program is an existing program that provides for detection of age related pressure boundary degradation of components exposed to air environments due to aging effects such as ***cracking***, loss of leaktightness, loss of material or loss of preload in various systems penetrating containment. The program also provides for detection of age related degradation in material properties of gaskets, o-rings, and packing materials for the primary containment pressure boundary access points.

Enclosure C

Salem Nuclear Generating Station, Units 1 and 2
License Renewal Commitment List Changes

This Enclosure includes an update to the Salem Nuclear Generating Station, Units 1 and 2, License Renewal Application (LRA) Appendix A, Section A.5 License Renewal Commitment List, as a result of the PSEG response to the following RAI:

RAI 3.5.2.2.1.7-01

Note: For clarity, portions of the original LRA License Renewal Commitment List text are repeated in this Enclosure. Added text is shown in ***Bold Italics***.

As a result of the response to **RAI 3.5.2.2.1.7-01** provided in Enclosure A of this letter, LRA Appendix A, Table A.5 License Renewal Commitment List, line Item 28 on page A-67, is revised to add enhancement 3 as shown below. The RAI that led to the commitment modification is listed in the "SOURCE" column adjacent to the beginning of the new text. Any other actions described in this submittal represent intended or planned actions. They are described for the NRC's information and are not regulatory commitments.

A.5 License Renewal Commitment List

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA APP. A)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
28	ASME Section XI, Subsection IWE	ASME Section XI, Subsection IWE is an existing program that will be enhanced to include: ***3. ASME Section XI, Subsection IWE program scope will be revised to include the following welds that are currently exempted from Subsection IWE and governed under ASME Section XI, Subsection IWB or IWC. The scope of the revision will include the cap plate to penetrating pipe pressure boundary welds, for penetrating pipe constructed of stainless steel for those penetrations with a normal operating temperature greater than 140 degrees F.***	A.2.1.28	Program to be enhanced prior to the period of extended operation.	***Salem letter LR-N10-0244 RAI 3.5.2.2.1.7-01***