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Protecting People and the Environment

NUREG-2101

Safety Evaluation Report

Related to the License Renewal of Salem Nuclear Generating Station

Docket Numbers 50-272 and 50-311

PSEG Nuclear, LLC

Office of Nuclear Reactor Regulation

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Office of Nuclear Reactor Regulation

ABSTRACT

This safety evaluation report (SER) documents the technical review of the Salem Nuclear Generating Station, Units 1 and 2, (Salem) license renewal application (LRA) by the U.S. Nuclear Regulatory Commission (NRC) staff (the staff). By letter dated August 18, 2009, PSEG Nuclear, LLC (PSEG or the applicant) submitted the LRA in accordance with Title 10, Part 54, of the *Code of Federal Regulations*, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants." PSEG requests renewal of the operating licenses (Facility Operating License Numbers DPR-70 and DPR-75) for a period of 20 years beyond the current expiration at midnight August 13, 2016, for Unit 1, and at midnight on April 18, 2020, for Unit 2.

Salem is located approximately 40 miles from Philadelphia, PA, and 8 miles from Salem, NJ. The NRC issued the construction permits for Unit 1 and Unit 2 on August 25, 1968. The NRC issued the operating license for Unit 1 on December 1, 1976, and for Unit 2 on May 20, 1981. Both units are pressurized water reactors that were designed and supplied by Westinghouse. License Amendment Nos. 243 (Salem Unit 1) and 224 (Salem Unit 2), dated May 25, 2001, authorized a 1.4 percent increase in the licensed rated power level of each unit to 3,459 megawatt thermal (MWt).

This SER presents the status of the staff's review of information submitted through May 18, 2011, the cutoff date for consideration in this SER. The staff has resolved all issues associated with requests for additional information and closed all open items since publishing the SER with Open Items. The staff did not identify any new open items that must be resolved before any final determination can be made on the LRA.

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1.7 Summary of Proposed License Conditions

Following the staff's review of the LRA, including subsequent information and clarifications provided by the applicant, the staff identified four proposed license conditions.

The first license condition requires the applicant to update the UFSAR supplement required by 10 CFR 54.21(d) in the UFSAR following the issuance of the renewed license.

The second license condition requires the applicant to complete the commitments in the UFSAR supplement and notify the NRC in writing when implementation of those activities required prior to the period of extended operation are complete and can be verified by NRC inspection.

The third license condition requires that all capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of ASTM E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage must be maintained for future insertion. Any changes to storage requirements must be approved by the NRC.

The fourth license condition requires the applicant to take one core sample in the Unit 1 SFP west wall, by the end of 2013, and one core sample in the east wall where there have been indications of borated water ingress through the concrete, by the end of 2015. The core samples (east and west walls) will expose the rebar, which will be examined for signs of corrosion. Any sample showing signs of concrete degradation and/or rebar corrosion will be entered into the licensee's corrective action program for further evaluation. The licensee shall submit a report in accordance with 10 CFR 50.4 no later than three months after each sample is taken on the results, recommendations, and any additional planned actions.

"Cracking Due to Stress Corrosion Cracking, Primary Water Stress Corrosion Cracking, and Irradiation-Assisted Stress Corrosion Cracking." Therefore, the staff issued RAI 3.1.2.2.12-1 requesting that the applicant 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.

Although the response, dated July 15, 2010, to RAI 3.1.2.2.12-1 did not provide direct justification, the staff determines that the proposed industry program for managing PWR internals as documented in MRP-227 is structured around components, not around aging effects. Therefore, not identifying PWSCC as an aging effect for certain components in LRA Table 3.1.2-3 has no impact on the AMP to be implemented for managing PWR internals. MRP-227 is currently under the NRC's review in a separate effort. Hence, RAI 3.1.2.2.12-1 is resolved.

Based on a review of the programs identified above, the staff concludes that the applicant's programs meet SRP-LR Section 3.1.2.2.12 criteria. For those line items that apply to LRA Section 3.1.2.2.12, the staff determines that the LRA is consistent with the GALL Report and that the applicant has demonstrated that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB during the period of extended operation, as required by 10 CFR 54.21(a)(3).

3.1.2.2.13 Cracking Due to Primary Water Stress-Corrosion Cracking

The staff reviewed LRA Section 3.1.2.2.13 against the criteria in SRP-LR Section 3.1.2.2.13, which recommends no further AMR if the applicant complies with applicable NRC orders and provides a commitment in the UFSAR supplement to implement applicable: (1) bulletins and GLs and (2) staff-accepted industry guidelines. The staff noted that the applicant's commitment (Commitment No. 46) in LRA Appendix A, Section A.5 commits to the implementation of the Nickel Alloy Aging Management Program and that various portions of that program contain language which is consistent with the commitment described in SRP-LR Section 3.1.2.2.13. The staff also notes that all of the AMR results lines that refer to Table 3.1.1, item 3.1.1-31 are aligned with the applicant's commitment as described in LRA Appendix A, Section A.5. The staff finds the applicant's proposal acceptable because the applicant provided the appropriate commitment in the UFSAR supplement and the AMR results lines refer to the commitment.

3.1.2.2.14 Wall Thinning Due to Flow-Accelerated Corrosion

LRA Section 3.1.2.2.14 refers to Table 3.1.1, item 3.1.1-32 and addresses the steel SG feedwater inlet ring and supports exposed to treated water, which are being managed for wall thinning due to flow-accelerated corrosion by the Steam Generator Tube Integrity Program. In LRA Table 3.1.1, item 3.1.1-32 and LRA Section 3.1.2.2.14, the applicant stated that the Steam Generator Tube Integrity Program will be used to manage wall thinning in the feedwater inlet ring and supports. The applicant further stated that the Steam Generator Tube Integrity Program implements a number of industry guidelines and incorporates a balance of prevention, inspection, evaluation, repair, and leakage monitoring measures to assure that existing environmental conditions are not causing wall thinning that could result in loss of component intended function.

The staff reviewed LRA Section 3.1.2.2.14 against the criteria in SRP-LR Section 3.1.2.2.14, which state that wall thinning due to flow-accelerated corrosion could occur in the steel

feedwater inlet rings and supports. The GALL Report refers to NRC IN 91-19, "Steam Generator Feedwater Distribution Piping Damage," for evidence of flow-accelerated corrosion in SGs and recommends that a plant-specific AMP be evaluated because existing programs may not be capable of mitigating or detecting wall thinning due to flow-accelerated corrosion.

The staff reviewed the applicant's Steam Generator Tube Integrity Program and its evaluation is documented in SER Section 3.0.3.1.8. In its review of components associated with LRA item 3.1.1-32, the staff noted that the GALL Report recommends that a plant-specific AMP be evaluated and the applicant credits the Steam Generator Tube Integrity Program to manage wall thinning in these components.

The staff noted that the Steam Generator Tube Integrity Program description in LRA Section B.2.1.10 states that the program includes managing the aging effect of wall thinning. However, the LRA does not describe what inspection or analytical techniques are used to ensure that excessive wall thinning in components does not occur.

By letter dated June 29, 2010, the staff issued RAI 3.1.2.2.14-01 requesting that the applicant describe its examination techniques and evaluation methodology used to manage wall thinning in the SG feedwater inlet rings and supports.

In its response to the RAI, dated July 28, 2010, the applicant stated that the Steam Generator Tube Integrity Program uses visual inspections of the SGs' secondary-side internals and that it does not include predictive analytical techniques. The applicant further stated that the Unit 1 SGs are Westinghouse Model F, with feedwater rings and supports constructed of carbon steel, and that the Unit 2 SGs are AREVA Model 61/19T, with feedwater ring supports constructed of low-alloy steel plates and feedwater rings constructed of 316L stainless steel. The applicant also stated that the aging effect and mechanism of wall thinning due to flow-accelerated corrosion does not apply to the stainless steel Unit 2 SG feedwater ring.

The applicant stated that the visual inspection techniques and associated acceptance criteria are determined by a SG degradation assessment which evaluates internal and external operating experience, industry guidance, design features, and materials of construction. The applicant stated that these inspections identify the general condition of the applicable SG components and inspect for evidence of erosion-corrosion, irregular geometry, and structural changes and that the acceptance criteria require that there be no visible signs of deterioration in the Unit 1 feedwater rings or in the Units 1 and 2 feedwater ring supports. The applicant further stated that it performs an operational assessment in accordance with NEI 97-06, "Steam Generator Program Guidelines," and applicable EPRI documents to confirm that acceptance criteria are met for the SGs to return to service and operate for the subsequent cycle and that the operational assessment ensures that deficiencies are identified and corrective actions are taken before loss of component intended function occurs. The applicant also stated that while preparing its response, it noted that LRA Table 3.1.2-4, "Summary of Aging Management Evaluations for SGs," did not correctly include the material differences between Unit 1 feedwater rings (carbon steel) and Unit 2 feedwater rings (stainless steel). The applicant revised this table to show that wall thinning due to flow-accelerated corrosion is applicable for the Unit 1 carbon steel feedwater rings and for the Units 1 and 2 carbon steel or low-alloy steel supports, but is not applicable for the Unit 2 stainless steel feedwater rings. In subsequent letters dated August 26, 2010, and October 8, 2010, the applicant further revised LRA Table 3.1.2-4 to state that its SG designs include both carbon steel and stainless steel secondary internals and that wall thinning due to flow-accelerated corrosion is also applicable for the carbon steel SG secondary internals. The applicant stated that these carbon steel and low-alloy steel

components are in a treated water environment (secondary feedwater/steam) and the aging effect will be managed by the Steam Generator Tube Integrity Program. The applicant also added lines showing that loss of material due to pitting and crevice corrosion and cracking due to SCC are aging effects applicable for the Unit 2 stainless steel feedwater rings. These stainless steel components are in a treated water environment and those aging effects will be managed by a combination of the Water Chemistry Program and the Steam Generator Tube Integrity Program.

In its review of the applicant's response, the staff noted that GALL AMP XI.M19, "Steam Generator Tube Integrity," references NEI 97-06. The staff determined that NEI 97-06 provides acceptable guidance for inspection and assessment of additional SG components, including the feedwater rings, supports, and secondary internals, consistent with the GALL Report. The staff further noted that industry operating experience supports the applicant's claim that flow-accelerated corrosion is not applicable to the Unit 2 stainless steel feedwater rings and the secondary internals. The staff finds the Steam Generator Tube Integrity Program acceptable to manage aging of the Unit 1 carbon steel feedwater rings and supports, the Unit 2 low-alloy steel feedwater ring supports, and the carbon steel secondary internals because the program: (1) provides visual inspections of the subject SG components based on recommendations of NEI 97-06, (2) includes assessments of inspection results against appropriate acceptance criteria, and (3) provides for corrective actions to be taken, as needed, to ensure that the subject components remain capable of performing their intended functions between scheduled SG inspections.

The staff finds the applicant's change to LRA Table 3.1.2-4 acceptable because these changes: (1) document the material difference between the steel feedwater rings in Unit 1 and the stainless steel feedwater rings in Unit 2 and (2) for the stainless steel feedwater rings and secondary internals, the staff has determined that the applicant's AMR results are acceptable as documented in SER Sections 3.4.2.2.6 and 3.4.2.2.7 for LRA Table 3.4.1, items 3.4.1-14 and 3.4.1-16, respectively.

Based on its review, the staff finds the applicant's response to RAI 3.1.2.2.14-01 acceptable as described above. The staff's concern described in RAI 3.1.2.2.14-01 is resolved.

Based on the program identified above, the staff concludes that the applicant's program meets SRP-LR Section 3.1.2.2.14 criteria. For those line items that apply to LRA Section 3.1.2.2.14, the staff determines that the LRA is consistent with the GALL Report and that the applicant has demonstrated that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB during the period of extended operation, as required by 10 CFR 54.21(a)(3).

3.1.2.2.15 Changes in Dimensions Due to Void Swelling

The staff reviewed LRA Section 3.1.2.2.15 against the criteria in SRP-LR Section 3.1.2.2.15. LRA Section 3.1.2.2.15 addresses changes in dimensions due to void swelling in stainless steel and nickel-alloy PWR reactor internal components exposed to reactor coolant as an aging effect that the applicant will manage, consistent with the SRP-LR, by the commitment of the PWR Vessel Internals Program.

SRP-LR Section 3.1.2.2.15 states that:

> Changes in dimensions due to void swelling could occur in stainless steel and nickel alloy PWR reactor internal components exposed to reactor coolant. The GALL Report recommends no further AMR if the applicant provides a commitment in the [U]FSAR Supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.

As described in LRA Section 3.1.2.2.15, the applicant made a commitment to incorporate all three GALL Report requirements stated above to manage this aging effect. The PWR Vessel Internals Program contains this commitment (Commitment No. 7). Commitment No. 7 is also identified in UFSAR Section A.2.1.7. Therefore, the staff concludes that the applicant's program meets the SRP-LR Section 3.1.2.2.15 criteria. The staff also examined LRA Table 3.1.2-3 to find out whether the RPV internals subjected to these aging effects are consistent with those listed in GALL Report Table IV.B2. The staff confirmed that LRA Table 3.1.2-3 identified all GALL Report Table IV.B2 items and the components under them for this aging effect (IV.B2-1, IV.B2-4, IV.B2-7, IV.B2-11, IV.B2-15, IV.B2-19, IV.B2-23, IV.B2-27, IV.B2-29, IV.B2-35, IV.B2-39, and IV.B2-41). For GALL Report items IV.B2-4, IV.B2-19, and IV.B2-29, the applicant identified additional RPV internal components which are different but consistent with these GALL Report items for material, environment, and aging effect. For most of the GALL Report Table IV.B2 items mentioned above, LRA Table 3.1.2-3 provides a set of subcomponents to represent a single component in GALL Report Table IV.B2. The applicant's approach of including additional components under the required AMP for GALL Report items IV.B2-4, IV.B2-19, and IV.B2-29 is acceptable.

Based on a review of the program identified above, the staff concludes that the applicant's program meets SRP-LR Section 3.1.2.2.15 criteria. For those line items that apply to LRA Section 3.1.2.2.15, the staff determines that the LRA is consistent with the GALL Report and that the applicant has demonstrated that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB during the period of extended operation, as required by 10 CFR 54.21(a)(3).

3.1.2.2.16 Cracking Due to Stress-Corrosion Cracking and Primary Water Stress-Corrosion Cracking

The staff reviewed LRA Section 3.1.2.2.16 against the criteria in SRP-LR Section 3.1.2.2.16.

(1) LRA Section 3.1.2.2.16.1 refers to Table 3.1.1, item 3.1.1-34, and addresses stainless steel and nickel-alloy reactor CRD head penetration pressure housings, which are managed for cracking due to SCC and PWSCC. The LRA states that the applicant will implement the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD and Water Chemistry programs to manage the cracking due to SCC in the stainless steel reactor CRD head penetration pressure housings.

    The staff reviewed LRA Section 3.1.2.2.16.1 against the criteria in SRP-LR Section 3.1.2.2.16.1, which state that cracking due to SCC could occur on the primary coolant side of PWR steel SG upper and lower heads, tubesheets, and tube-to-tubesheet welds made or clad with stainless steel. The SRP-LR also states that cracking due to PWSCC could occur on the primary coolant side of PWR steel SG upper

and lower heads, tubesheets, and tube-to-tubesheet welds made or clad with nickel alloy. The staff noted that the GALL Report recommends the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD and Water Chemistry programs to manage these aging effects. In addition, the GALL Report indicates that no further AMR of nickel alloys are required if the applicant complies with applicable NRC orders and provides a commitment in the UFSAR supplement to implement applicable NRC bulletins, GLs, and NRC staff-accepted industry guidelines.

The staff further reviewed the LRA and identified in Table 3.1.1, item 3.1.1-34, and Table 3.1.2-2 that the applicant addressed SCC of stainless steel reactor CRD head penetration pressure housings exposed to reactor coolant and credited the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD and Water Chemistry programs to manage the aging effect. The staff reviewed the applicant's ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD and Water Chemistry programs and its evaluations are documented in SER Sections 3.0.3.1.1 and 3.0.3.1.2, respectively. In its review, the staff finds that the credited programs are adequate to manage the aging effect because: (a) the Water Chemistry Program monitors the plant water chemistry parameters against the established parameter limits and, if a parameter exceeds the limit, the program performs adequate actions such that the water chemistry control continues to mitigate the aging effect; (b) the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program includes inspections of selected components to verify the effectiveness of the Water Chemistry Program consistent with the GALL Report; and (c) the inspections in accordance with ASME Code Section XI can ensure that significant degradation does not occur and the intended function of the component is maintained during the period of extended operation consistent with the GALL Report.

In LRA Table 3.1.1, the applicant further stated that item 3.1.1-35 is not applicable because Salem Units 1 and 2 SGs are not a once-through design and, therefore, do not have the components associated with this model of SGs. The staff noted that the GALL Report, Revision 1, Volume 2 indicates that item 3.1.1-35 is only applicable to OTSGs, but not to recirculating SGs.

UFSAR Section 5.5.2.2.2 describes Unit 1 Model F SG tubes as fabricated from Alloy 600TT and welded to the Inconel cladding on the primary face of the tube plate. UFSAR Section 5.5.2.2.1 describes Unit 2 replacement SG tubes as fabricated from Alloy 690TT and weld clad with Alloy 600 at the primary side of the tubesheet.

The staff noted that ASME Code Section XI does not require any inspection of the tube-to-tubesheet welds. In addition, no specific NRC orders or bulletins require examination of this weld. However, the staff's concern is that, if the tubesheet cladding is Alloy 600, the autogenous tube-to-tubesheet weld may not have sufficient Chromium content to prevent initiation of PWSCC, even when the SG tubes are made from Alloy 690TT, as it is the configuration for the applicant's Unit 2 SG tubes. Consequently, such a PWSCC crack initiated in this region, close to a tube, could propagate into/through the weld, causing a failure of the weld and of the RCPB, even for recirculating SGs such as those for both units. Therefore, unless the NRC has approved a redefinition of the pressure boundary in which the autogenous tube-to-tubesheet weld is no longer included, or the tubesheet cladding and welds are not susceptible to PWSCC, the staff considers that the effectiveness of the primary water chemistry program should be verified to ensure PWSCC does not occur.

By letter dated November 4, 2010, the staff issued RAI 3.1.1-03 requesting that the applicant clarify for Unit 1 SGs whether the tube-to-tubesheet welds are included in the RCPB or if alternate repair criteria (ARC) have been permanently approved. Furthermore, the staff noted that if there is no ARC permanently approved, the applicant should provide a plant-specific AMP that will complement the primary water chemistry program in order to verify the effectiveness of the primary water chemistry program and ensure that cracking due to PWSCC is not occurring in tube-to-tubesheet welds. For Unit 2 SGs tube-to-tubesheet welds, the staff requested that the applicant provide either a plant-specific AMP that will complement the primary water chemistry program, in order to verify the effectiveness of the primary water chemistry program and ensure that cracking due to PWSCC is not occurring in tube-to-tubesheet welds, or provide a rationale for why such a program is not needed. The staff identified this as Open Item OI 3.1.2.2.16-1.

In its response dated December 1, 2010, and revised by its letter dated December 15, 2010, the applicant committed to the following:

> [It] will develop a plan for each Unit to address the potential for cracking of the primary to secondary pressure boundary due to PWSCC of tube-to-tubesheet welds. Each plan will consist of two options.

The applicant committed in Commitment No. 51 to develop a plan for each unit before the period of extended operation. Each plan consists of two options.

For Unit 1, the applicant stated that the TSs were amended on March 29, 2010 (ADAMS Accession No. ML100570452), approving a one-time change to TS Section 6.8.4.i, “Steam Generator Program.” The applicant explained that this amendment is an approval for ARC and limits the required inspection (and repair if degradation is found) to the portions of the SG tubes passing through the upper 13.1 inches of the approximate 21-inch tubesheet region; therefore, the bottom 7.9 inches of the tube, including the tube-to-tubesheet weld, are not presently considered part of the RCPB. The applicant further stated that the TS amendment, used in the spring 2010 refueling outage, is valid until the next scheduled SG tube inspections presently scheduled for the spring 2013 refueling outage. Since this ARC approval expires by the spring 2013 refueling outage, which is prior to the Unit 1 period of extended operation, the applicant stated that it would develop a plan to address potential cracking of the SG primary to secondary pressure boundary due to PWSCC of tube-to-tubesheet welds consisting of the following two options:

In the first option, the applicant stated that it would request permanent NRC approval for ARC, which re-defines the RCPB to no longer include the autogenous tube-to-tubesheet welds prior to the Unit 1 period of extended operation. The applicant further stated that if permanent approval for ARC has not been granted by the NRC prior to Unit 1 entering its period of extended operation, it would implement the second option.

In the second option, the applicant stated that it would perform a one-time inspection of a representative number of tube-to-tubesheet welds in each of the four SGs to determine if PWSCC is present and verify the effectiveness of the Water Chemistry Program. The applicant also stated that if weld cracking is identified, the condition would be resolved through repair or engineering evaluation to justify continued service, as appropriate, and a periodic monitoring program would be established to perform routine tube-to-tubesheet inspections for the remaining life of the SGs.

Moreover, the applicant stated that the SG tube-to-tubesheet welds have been in service for approximately 12 years since the Unit 1 SGs were replaced in April 1998. The applicant further stated that these inspections would be performed between April 2018 and April 2023, such that the SGs will have been in service between 20 and 25 years.

For Unit 2, the applicant stated that the plan would also address potential failure of the SG RCPB due to PWSCC of tube-to-tubesheet welds and would consist of the following two options:

In the first option, the applicant stated that it would perform an analytical evaluation of the SG tube-to-tubesheet welds in order to establish a technical basis for either determining that the tubesheet cladding and welds are not susceptible to PWSCC, or redefining the pressure boundary such that the autogenous tube-to-tubesheet weld is no longer included and, therefore, not required for the RCPB function. The applicant further stated that the redefinition of the RCPB would be submitted as part of a license amendment request requiring approval from the NRC, and the approved analytical evaluation would supersede the need to develop a plant-specific AMP to verify the effectiveness of the Water Chemistry Program.

In the second option, the applicant stated that it would perform a one-time inspection of a representative number of tube-to-tubesheet welds in each of the four SGs to determine if PWSCC is present and verify the effectiveness of the Water Chemistry Program. The applicant also stated that if weld cracking is identified, the condition would be resolved through repair or engineering evaluation to justify continued service, as appropriate, and a periodic monitoring program would be established to perform routine tube-to-tubesheet inspections for the remaining life of the SGs.

Moreover, the applicant stated that the SG tube-to-tubesheet welds for Unit 2 have been in service for less than 3 years since the SGs had been replaced in April 2008. The applicant further stated that these inspections would be performed between April 2028 and April 2033, such that the SGs will have been in service between 20 and 25 years.

Based on its review, the staff finds the applicant‘s response to RAI 3.1.1-03 and associated Commitment No. 51 acceptable because the applicant will manage the aging effect of cracking due to PWSCC in the SG tube-to-tubesheet welds either by demonstrating that those welds are no longer required for the SG RCPB function (or not susceptible to PWSCC for Unit 2), or by implementing a one-time inspection on a representative number of tube-to-tubesheet welds of each SG to determine if PWSCC is present, in a time period consistent with the detection of potential PWSCC cracks. The staff finds that the timing of this inspection for each unit is acceptable because at the time of the inspections, the respective SGs will have been in operation for between 20 and 25 years, and it is unlikely that significant detrimental PWSCC cracking will have initiated. The staff also noted that, in case the aging effect is identified, this one-time inspection would be accompanied by corrective actions, including an evaluation of the degradation and the implementation of routine inspections of the tube-to-tubesheet welds for the remaining life of the SGs. The staff’s concern described in RAI 3.1.1-03 is resolved, and Open item OI 3.1.2.2.16-1 is closed.

The staff concludes that the applicant has demonstrated that the effects of aging for these components will be adequately managed so that their intended function(s) will be maintained consistent with the CLB during the period of extended operation, as required by 10 CFR 54.21(a)(3).

(2) LRA Section 3.1.2.2.16.2 refers to Table 3.1.1, item 3.1.1-36 and addresses the SCC in the stainless steel pressurizer spray head exposed to reactor coolant. The LRA further states that it will implement the Water Chemistry Program and One-Time Inspection Program to manage the aging effect.

The staff reviewed LRA Section 3.1.2.2.16.2 against the criteria in SRP-LR Section 3.1.2.2.16.2, which state that cracking due to SCC could occur on stainless steel pressurizer spray heads. The SRP-LR also states that the existing program relies on control of water chemistry to mitigate this aging effect. The SRP-LR further states that the GALL Report recommends a one-time inspection to confirm that the cracking does not occur. The staff also noted that the GALL Report, item IV.C2-17, recommends the Water Chemistry Program and the One-Time Inspection Program to manage the aging effect of stainless steel components. The staff noted that the GALL Report recommends the One-Time Inspection Program to verify the effectiveness of the water chemistry control program.

The staff reviewed the LRA and identified in Table 3.1.1, item 3.1.1-36, and Table 3.1.2-1 that the applicant credited the Water Chemistry and One-Time Inspection programs to manage the SCC in the stainless steel pressurizer spray head exposed to reactor coolant. The staff also reviewed the applicant's Water Chemistry and One-Time Inspection programs and its evaluations are documented in SER Sections 3.0.3.1.2 and 3.0.3.1.11, respectively. The applicant indicated that the One-Time Inspection Program includes a one-time inspection of more susceptible materials in potentially more aggressive environments to manage the aging effect. The staff finds that the credited programs are adequate to manage the aging effect because: (a) the Water Chemistry Program monitors the plant water chemistry control parameters against the established parameter limits and, if a parameter exceeds the limit, the program performs adequate actions such that the water chemistry control continues to mitigate the aging effect; (b) the One-Time Inspection Program includes a one-time inspection of selected components to verify the effectiveness of the Water Chemistry Program consistent with the GALL Report; and (c) the one-time inspection can ensure that significant degradation does not occur and the component's intended function is maintained during the period of extended operation. On the basis of its review, the staff finds that the applicant's AMR results are consistent with those under GALL Report, Volume 2, item IV.C2-17 and the applicant satisfied the acceptance criteria in SRP-LR Section 3.1.2.2.16.2.

Based on the programs identified, the staff concludes that the applicant's programs meet SRP-LR Section 3.1.2.2.16 criteria. For those items that apply to LRA Section 3.1.2.2.16, the staff determines that the LRA is consistent with the GALL Report and that the applicant has demonstrated that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB during the period of extended operation, as required by 10 CFR 54.21(a)(3).

3.1.2.2.17 Cracking Due to Stress-Corrosion Cracking, Primary Water Stress-Corrosion Cracking, and Irradiation-Assisted Stress-Corrosion Cracking

The staff reviewed LRA Section 3.1.2.2.17 against the criteria in SRP-LR Section 3.1.2.2.17. LRA Section 3.1.2.2.17 addresses cracking due to SCC, PWSCC, and IASCC in stainless steel and nickel-alloy PWR reactor internal components exposed to reactor coolant and neutron flux as an aging effect that the applicant will manage, consistent with the SRP-LR, with the Water Chemistry Program and the commitment of the PWR Vessel Internals Program.

SRP-LR Section 3.1.2.2.17 states that:

> Cracking due to SCC, PWSCC, and IASCC could occur in PWR stainless steel and nickel alloy reactor vessel internals components. The existing program relies on control of water chemistry to mitigate these effects. However, the existing program should be augmented to manage these aging effects for reactor vessel internals components. The GALL Report recommends no further AMR if the applicant provides a commitment in the [U]FSAR Supplement to (1) participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.

As indicated in SER Section 3.0.3.1.2, the staff accepts the Water Chemistry Program for mitigating the aging effects due to SCC, PWSCC, and IASCC, meeting one of the requirements mentioned in SRP-LR Section 3.1.2.2.17. Furthermore, the applicant made a commitment to incorporate all three GALL Report requirements stated above to manage this aging effect. The PWR Vessel Internals Program contains this commitment (Commitment No. 7). Commitment No. 7 is also identified in UFSAR Section A.2.1.7. Therefore, the staff concludes that the applicant’s program meets the SRP-LR Section 3.1.2.2.17 criteria. The staff also confirmed that LRA Table 3.1.2-3 identified all GALL Report Table IV.B2 items and the components under them for this aging effect (IV.B2-16, IV.B2-20, IV.B2-28, and IV.B2-40). For GALL Report item IV.B2-20, the applicant identified additional RPV internal components which are different but consistent with these GALL Report items for material, environment, and aging effect. For most of the GALL Report Table IV.B2 items mentioned above, LRA Table 3.1.2-3 provides a set of subcomponents to represent a single component in GALL Report Table IV.B2. The applicant’s approach of including additional components under the required AMP for GALL Report item IV.B2-20 is conservative and acceptable.

It was mentioned in SER Section 3.1.2.2.12 that LRA Table 3.1.2-3 does not distinguish the aging effects discussed in LRA Sections 3.1.1.1.12 and 3.1.2.2.17. This has no impact on the AMP managing the PWR internals under these two aging effects as explained in SER Section 3.1.2.2.12.

Based on a review of the programs identified, the staff concludes that the applicant’s programs meet SRP-LR Section 3.1.2.2.17 criteria. For those line items that apply to LRA Section 3.1.2.2.17, the staff determines that the LRA is consistent with the GALL Report and that the applicant has demonstrated that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB during the period of extended operation, as required by 10 CFR 54.21(a)(3).

3.1.2.2.18 Quality Assurance for Aging Management of Nonsafety-Related Components

SER Section 3.0.4 provides the staff’s evaluation of the applicant’s QA program.

APPENDIX A

SALEM NUCLEAR GENERATING STATION LICENSE RENEWAL COMMITMENTS

During the review of the Salem Nuclear Generating Station (Salem) license renewal application (LRA) by the staff of the U.S. Nuclear Regulatory Commission (NRC or the staff), PSEG Nuclear, LLC (PSEG or the applicant) made commitments related to aging management programs (AMPs) to manage aging effects for structures and components. The following table lists these commitments along with the implementation schedules and sources for each commitment.

APPENDIX A: SALEM UNITS 1 AND 2 LICENSE RENEWAL COMMITMENTS

Item Number	Commitment	Updated Final Safety Analysis Report (UFSAR) Supplement Section/ LRA Section	Enhancement or Implementation Schedule	Source
1	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	A.2.1.1	Ongoing	LRA Section B.2.1.1
2	Water Chemistry	A.2.1.2	Ongoing	LRA Section B.2.1.2
3	Reactor Head Closure Studs	A.2.1.3	Ongoing	LRA Section B.2.1.3
4	Boric Acid Corrosion	A.2.1.4	Ongoing	LRA Section B.2.1.4
5	Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors	A.2.1.5	Ongoing	LRA Section B.2.1.5
6	Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) is a new program that will provide for aging management of the thermal embrittlement of CASS piping, piping elements and piping components in a reactor coolant environment. The program will include a screening for components susceptible to thermal aging embrittlement based on casting method, molybdenum content, and percent ferrite. For "potentially susceptible" components, thermal aging embrittlement will be managed through either an enhanced volumetric inspection or a component-specific flaw tolerance evaluation.	A.2.1.6	Program to be implemented prior to the period of extended operation.	LRA Section B.2.1.6
7	PWR Vessel Internals is a new program that will include the following activities: 1. Participate in the industry programs for investigating and managing aging effects on reactor internals. 2. Evaluate and implement the results of the industry programs as applicable to the reactor internals. 3. Upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.	A.2.1.7	Program to be implemented prior to the period of extended operation. Inspection plan to be submitted to NRC not less than 24 months prior to the period of extended operation.	LRA Section B.2.1.7

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Item Number	Commitment	Updated Final Safety Analysis Report (UFSAR) Supplement Section/ LRA Section	Enhancement or Implementation Schedule	Source
8	Flow-Accelerated Corrosion	A.2.1.8	Ongoing	LRA Section B.2.1.8
9	Bolting Integrity Program is an existing program that will be enhanced to include: 1. In the following cases, bolting material should not be reused: a. Galvanized bolts and nuts, b. ASTM A490 bolts; and c. Any bolt and nut tightened by the turn of nut method.	A.2.1.9	Program to be enhanced prior to the period of extended operation.	LRA Section B.2.1.9
10	Steam Generator Tube Integrity	A.2.1.10	Ongoing	LRA Section B.2.1.10
11	Open-Cycle Cooling Water System	A.2.1.11	Ongoing	LRA Section B.2.1.11
12	Closed-Cycle Cooling Water System is an existing program that will be enhanced to include: 1. The Component Cooling System is not currently analyzed for sulfates, which is not consistent with the EPRI standard. The program will be enhanced to include monitoring this parameter as part of the Closed-Cycle Cooling Water program. 2. The emergency diesel generator jacket water system is not currently analyzed for azole or ammonia, chlorides, fluorides, and microbiologically-influenced corrosion in accordance with the current EPRI standard. The program will be enhanced to include these parameters as part of the Closed-Cycle Cooling Water program. 3. The Closed-Cycle Cooling Water program for the Chilled Water System will have a program or hardware change to bring the system chemistry parameters into compliance with EPRI 1007820, prior to the period of extended operation. 4. New recurring tasks will be established to enhance the performance monitoring of selected heat exchangers cooled by Component Cooling System. 5. New recurring tasks will be established for enhancing the	A.2.1.12	Program to be enhanced and one-time inspections to be implemented prior to the period of extended operation.	LRA Section B.2.1.12

APPENDIX A: SALEM UNITS 1 AND 2 LICENSE RENEWAL COMMITMENTS

Item Number	Commitment	Updated Final Safety Analysis Report (UFSAR) Supplement Section/ LRA Section	Enhancement or Implementation Schedule	Source
	performance monitoring of selected Chilled Water System components. 6. A one-time inspection of selected components will be established for Chilled Water System piping to confirm the effectiveness of the Closed-Cycle Cooling Water program. 7. A one-time inspection of selected closed-cycle cooling water components in stagnant flow areas will be conducted to confirm the effectiveness of the Closed-Cycle Cooling Water program. 8. A one-time inspection of selected closed-cycle cooling water chemical mixing tanks and associated piping will be conducted to confirm the effectiveness of the closed cycle cooling water program on the interior surfaces of the tanks and associated piping. 9. The program will be enhanced such that the Heating Water and Heating Steam System will have a pure water control program instituted, in accordance with EPRI 1007820, prior to the period of extended operation. 10. New recurring tasks will be established for enhancing the performance monitoring of selected Heating Water and Heating Steam System components. 11. A one-time inspection of selected Heating Water and Heating Steam System piping will be conducted to confirm the effectiveness of the Closed-Cycle Cooling Water program.			
13	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems is an existing program that will be enhanced to include: 1. Visual inspection of structural components and structural bolts for loss of material due to general, pitting, and crevice corrosion and structural bolting for loss of preload due to self-loosening. 2. Visual inspection of the rails in the rail system for loss of material due to wear.	A.2.1.13	Program to be enhanced prior to the period of extended operation.	LRA Section B.2.1.13

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Item Number	**Commitment**	**Updated Final Safety Analysis Report (UFSAR) Supplement Section/ LRA Section**	**Enhancement or Implementation Schedule**	**Source**
	3. The acceptance criteria will be enhanced to require evaluation of significant loss of material due to corrosion for structural components and structural bolts, and significant loss of material due to wear of rail in the rail system.			
14	Compressed Air Monitoring	A.2.1.14	Ongoing	LRA Section B.2.1.14
15	Fire Protection is an existing program that will be enhanced to include: 1. The routine inspection procedures will be enhanced to provide additional inspection guidance to identify degradation of fire barrier walls, ceilings, and floors for aging effects such as cracking, spalling and loss of material caused by freeze-thaw, chemical attack, and reaction with aggregates. 2. The fire pump supply line functional tests will be enhanced to provide specific guidance for examining exposed external surfaces of the fire pump diesel fuel oil supply line for corrosion during pump tests. 3. The Halon and Carbon Dioxide fire suppression system functional test procedures will be enhanced to include visual inspection of system piping and component external surfaces for signs of corrosion or other age related degradation, and for mechanical damage. The system functional test procedures will also be enhanced to include acceptance criteria stating that identified corrosion or mechanical damage will be evaluated with corrective action taken as appropriate.	A.2.1.15	Program to be enhanced prior to the period of extended operation.	LRA Section B.2.1.15 Salem Letter LR-N10-0225 RAI B.2.1.15-02 July 8, 2010
16	Fire Water System is an existing program that will be enhanced to include: 1. The Fire Water System aging management program will be enhanced to inspect selected portions of the water based fire protection system piping located aboveground and exposed to the fire water internal environment by non-intrusive volumetric examinations. These inspections shall be performed prior to the period of extended operation and will be performed every 10 years thereafter.	A.2.1.16	Program to be enhanced prior to the period of extended operation. Inspection schedule identified in Commitment.	LRA Section B.2.1.16

APPENDIX A: SALEM UNITS 1 AND 2 LICENSE RENEWAL COMMITMENTS

Item Number	Commitment	Updated Final Safety Analysis Report (UFSAR) Supplement Section/ LRA Section	Enhancement or Implementation Schedule	Source
	2. The Fire Water System aging management program will be enhanced to replace or perform 50-year sprinkler head inspections and testing using the guidance of NFPA-25 “Standard for the Inspection, Testing and Maintenance of Water-Based Fire Protection Systems” (2002 Edition), Section 5-3.1.1. These inspections will be performed by the 50-year in-service date and every 10-years thereafter.			
17	Aboveground Steel Tanks is an existing program that will be enhanced to include: 1. The program will be enhanced to include UT measurements of the bottom of the tanks that are supported on concrete foundations (Fire Protection Water Storage Tanks). Measured wall thickness will be monitored and trended if significant material loss is detected. These thickness measurements of the tank bottom will be taken and evaluated against design thickness and corrosion allowance to ensure that significant degradation is not occurring and the component intended function would be maintained during the extended period of operation. 2. The program will be enhanced to provide routine visual inspections of the Fire Protection Water Storage Tanks external surfaces. The visual inspection activities will include inspection of the grout or sealant between the tank bottom and the concrete foundation for signs of degradation.	A.2.1.17	Program to be enhanced prior to the period of extended operation. Tank bottom UT inspections will also be performed prior to the period of extended operation.	LRA Section B.2.1.17
18	Fuel Oil Chemistry is an existing program that will be enhanced to include: 1. Equivalent requirements for fuel oil purity and fuel oil testing as described by the Standard Technical Specifications. 2. Analysis for particulate contamination in new and stored fuel oil. 3. Addition of biocides, stabilizers and corrosion inhibitors as determined by fuel oil sampling or inspection activities.	A.2.1.18	Program to be enhanced and one-time inspections to be implemented prior to the period of extended operation.	LRA Section B.2.1.18

APPENDIX A: SALEM UNITS 1 AND 2 LICENSE RENEWAL COMMITMENTS

Item Number	Commitment	Updated Final Safety Analysis Report (UFSAR) Supplement Section/ LRA Section	Enhancement or Implementation Schedule	Source
	4. Quarterly analysis for bacteria in new and stored fuel oil. 5. Internal inspection of 350-gallon Fire Pump Day Tanks (S1DF-1DFE21 and S1DF-1DFE23) using visual inspections and ultrasonic thickness examination of tank bottoms. 6. Sampling of new fuel oil deliveries for API gravity and flash point prior to off load. 7. Internal inspection of the 30,000-gallon Fuel Oil Storage Tanks (S1DF-1DFE1, S1DF-1DFE2, S2DF-2DFE1 and S2DF-2DFE2) using visual inspections and ultrasonic thickness examination of tank bottoms. 8. To confirm the absence of any significant aging effects, a one-time inspection of each of the 550-gallon Diesel Fuel Oil Day Tanks will be performed.			
19	Reactor Vessel Surveillance is an existing program that will be enhanced to include: 1. The Reactor Vessel Surveillance program will be enhanced to state the bounding vessel inlet temperature (cold leg) limits and fluence projections, and to provide instructions for changes. a. Inlet Temperature Range Limitation: 525°F (min) to 590°F (max) b. Fluence Limitation (max.): 1.00×10^{20} n/cm^2 (E > 1.0 MeV) 2. The Reactor Vessel Surveillance program will be enhanced to describe the capsule storage requirements and the need to retain future pulled capsules. 3. The Reactor Vessel Surveillance program will be enhanced to specify a scheduled date for withdrawal of capsules including pulling one of the remaining four capsules during the period of extended operation to monitor the effects of long-term exposure to neutron embrittlement for each Salem Unit. Those dates shall be approved by the NRC prior to withdrawal of the capsules, in accordance with 10 CFR Part 50,	A.2.1.19	Program to be enhanced prior to the period of extended operation.	LRA Section B.2.1.19

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Item Number	**Commitment**	**Updated Final Safety Analysis Report (UFSAR) Supplement Section/ LRA Section**	**Enhancement or Implementation Schedule**	**Source**
	Appendix H. 4. The Reactor Vessel Surveillance program will be enhanced to incorporate the requirements for (1) withdrawing the remaining capsules when the monitor capsule is withdrawn during the period of extended operation and placing them in storage for the purpose of reinstituting the Reactor Vessel Surveillance Program if required, i.e. if the reactor vessel exposure conditions (neutron flux, spectrum, irradiation temperature, etc.) are altered, and subsequently the basis for the projection to 60 years warrant the reinstitution, and (2) changes to the reactor vessel exposure conditions and the potential need to re-institute a vessel surveillance program will be discussed with the NRC staff prior to changing the plant's licensing basis. 5. Enhancements to the current Reactor Vessel Surveillance program will be made to require that if future plant operations exceed the limitations or bounds specified for cold leg temperatures (vessel inlet) or higher fluence projections, then the impact of plant operation changes on the extent of reactor vessel embrittlement will be evaluated and the NRC shall be notified. a. Inlet Temperature Range Limitation: 525°F (min) to 590°F (max) b. Fluence Limitation (max.): 1.00 x 10^{20} n/cm^2 (E > 1.0 MeV)			
20	One-Time Inspection is a new program and will be used for the following: 1. To confirm the effectiveness of the Water Chemistry program to manage the loss of material, cracking, and the reduction of heat transfer aging effects for aluminum, copper alloy, nickel alloy, steel, stainless steel, and cast austenitic stainless steel in treated water, treated borated water where dissolved oxygen may not be controlled to less than 100 ppb, steam, and reactor coolant environments. 2. To confirm the effectiveness of the Fuel Oil Chemistry program to manage the loss of material aging effect for aluminum, copper alloy,	A.2.1.20	Program to be implemented prior to the period of extended operation. One-time inspections to be performed within the ten-year period prior to the period of extended operation.	LRA Section B.2.1.20 Salem Letter LR-N11-0005 RAI B.2.1.20-01 January 6, 2011 Salem Letter LR-N11-0148

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Item Number	**Commitment**	**Updated Final Safety Analysis Report (UFSAR) Supplement Section/ LRA Section**	**Enhancement or Implementation Schedule**	**Source**
	gray cast iron, steel and stainless steel in a fuel oil environment. 3. To confirm the effectiveness of the Lubricating Oil Analysis program to manage the loss of material and the reduction of heat transfer aging effects for aluminum, copper alloy, ductile cast iron, gray cast iron, steel, stainless steel, cast austenitic stainless steel and titanium alloy in a lubricating oil environment. The sample plan for inspections associated with the One-Time Inspection program will be developed to ensure there are adequate inspections to address each of the material, environment, and aging effect combinations. A sample size of 20% of the population (up to a maximum of 25 inspections) will be established for each of the sample groups.			May 18, 2011
21	Selective Leaching of Materials is a new program that will include one-time inspections of a representative sample of susceptible components to determine where loss of material due to selective leaching is occurring. A sample size of 20% of susceptible components will be subjected to a one-time inspection with a maximum of 25 inspections for each of the susceptible material groups. Where selective leaching is identified, further aging management activities will be implemented such that the component intended function is maintained consistent with the current licensing basis through the period of extended operation.	A.2.1.21	Program to be implemented prior to the period of extended operation. One-time inspections to be performed within the ten-year period prior to the period of extended operation.	LRA Section B.2.1.21 Salem Letter LR-N10-0324 September 1, 2010 Salem Letter LR-N11-0005 RAI B.2.1.21-01 January 6, 2011
22	Buried Piping Inspection is an existing program that will be enhanced to include: 1. A cathodic protection study will be performed prior to entering the period of extended operation to assess the possibility and benefits of installing a system, versus other mitigative and preventive actions. 2. A soil characterization study will be performed prior to entering the period of extended operation to determine soil corrosivity in the	A.2.1.22	Program to be enhanced prior to the period of extended operation. Inspection Schedule identified in Commitment.	LRA Section B.2.1.22 Salem Letter LR-N10-0322 RAI B.2.1.22 September 7, 2010 Salem Letter

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	vicinity of buried piping. The results of the study will be used as an input to the program so that inspections will be performed at the locations of highest risk. 3. At least one (1) opportunistic or focused excavation and inspection will be performed on each of the Fire Protection System material groupings, which include carbon steel, ductile cast iron, and gray cast iron piping and components during each ten (10) year period, beginning ten (10) years prior to entry into the period of extended operation. 4. The following inspections apply to buried, carbon steel, safety-related portions of the specified systems. A different segment for each system will be inspected in each ten year period. a. At least one (1) opportunistic or focused excavation and inspection on each of the Auxiliary Feedwater and Compressed Air systems during the ten (10) years prior to entering the period of extended operation. b. At least three (3) opportunistic or focused excavations and inspections of the Service Water System during the ten (10) years prior to entering the period of extended operation. c. If, as a result of the soil characterization study, it is determined that the soil is not corrosive in the vicinity of all of the Auxiliary Feedwater, Service Water, and Compressed Air systems, Salem will perform at least (1) opportunistic or focused excavation and inspection on each of the respective systems every ten (10) years during the period of extended operation. d. If, as a result of the soil characterization study, it is determined that the soil is corrosive in the vicinity of the Auxiliary Feedwater, Service Water, or Compressed Air systems, Salem will perform at least two (2) opportunistic or focused excavations and inspections on the respective			LR-N10-0372 RAI B.2.1.22-02 November 10, 2010 Salem Letter LR-N10-0444 RAI B.2.1.22-03 January 18, 2011

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Item Number	**Commitment**	**Updated Final Safety Analysis Report (UFSAR) Supplement Section/ LRA Section**	**Enhancement or Implementation Schedule**	**Source**
	susceptible system(s) every ten (10) years during the period of extended operation. 5. If, based on the results of the initial soil characterization study, it is determined that the soil is not corrosive in the vicinity of the Auxiliary Feedwater, Service Water, or Compressed Air systems, Salem will perform a second Soil Characterization Study within approximately fifteen (15) years of the original study. The results of the second soil study will be entered into the Corrective Action Program for evaluation. 6. The buried Auxiliary Feedwater System piping located inside the Unit 2 Fuel Transfer Tube Area (approximately 125 feet) will be replaced and rerouted above ground prior to entering the period of extended operation.			

APPENDIX A: SALEM UNITS 1 AND 2 LICENSE RENEWAL COMMITMENTS

Item Number	Commitment	Updated Final Safety Analysis Report (UFSAR) Supplement Section/ LRA Section	Enhancement or Implementation Schedule	Source
23	One-Time Inspection of ASME Code Class 1 Small-Bore Piping is a new program that will manage the aging effect of cracking in stainless steel small-bore, less than nominal pipe size (NPS) 4 inches and greater than or equal to NPS 1 Class 1 piping through the use of a combination of volumetric examinations and visual inspections. The One-Time Inspection of ASME Code Class 1 Small Bore-Piping is a new program that will be enhanced to include the following activity: Salem Units 1 and 2 will perform four volumetric examinations, two per unit, from a population of 36 susceptible Class 1 small-bore socket welds on Unit 1 and 34 susceptible Class 1 small-bore socket welds on Unit 2. Provided the technology is available, these inspections shall be performed prior to entering the period of extended operation. More specifically, the volumetric examinations will analyze Class 1 small-bore socket welds as follows: • Two Class 1 small-bore socket welds (one per unit) for intergranular stress corrosion cracking; and • Two Class 1 small-bore socket welds (one per unit) for cracking caused by thermal fatigue (thermal and mechanical loading)	A.2.1.23	Program to be implemented prior to the period of extended operation. One-time inspections to be performed within the ten-year period prior to the period of extended operation. Program to be enhanced prior to the period of extended operation. The inspection schedule will be consistent with the Salem ISI Program requirements.	LRA Section B.2.1.23 Salem Letter LR-N10-0247 RAI B.2.1.23-01 July 8, 2010
24	External Surfaces Monitoring is a new program that directs visual inspections of components such as piping, piping components, ducting and other components in the scope of license renewal, exposed to an air environment, to manage aging effects.	A.2.1.24	Program to be implemented prior to the period of extended operation.	LRA Section B.2.1.24
25	Flux Thimble Tube Inspection is a new program that manages the loss of material due to wear of the flux thimble tube materials using inspection methods such as eddy current testing.	A.2.1.25	Program to be implemented prior to the period of extended operation.	LRA Section B.2.1.25
26	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting	A.2.1.26	Program to be	LRA Section B.2.1.26

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Item Number	Commitment	Updated Final Safety Analysis Report (UFSAR) Supplement Section/ LRA Section	Enhancement or Implementation Schedule	Source
	Components is a new program that manages the aging of the internal surfaces of piping, piping components, piping elements, ducting components, tanks and heat exchanger components.		implemented prior to the period of extended operation.	
27	Lubricating Oil Analysis	A.2.1.27	Ongoing	LRA Section B.2.1.27
28	ASME Section XI, Subsection IWE is an existing program that will be enhanced to include: 1. Inspection of a sample of the inaccessible liner covered by insulation and lagging once prior to the period of extended operation and every 10 years thereafter. Should unacceptable degradation be found additional insulation will be removed as necessary to determine extent of condition in accordance with the corrective action process. Prior to the period of extended operation • The samples shall include 57 randomly selected containment liner insulation panels per unit. • The randomly selected containment liner insulation panels will not include containment liner insulation panels previously removed to allow for inspection. • The examination will be performed by either removing the containment liner insulation panels and performing a visual inspection, or by using a pulsed eddy current (PEC) remote inspection, with the containment liner insulation left in place, to detect evidence of loss of material. If evidence of loss of material is detected using PEC, the containment liner insulation panel will be subsequently removed to allow for visual and UT examinations. • All inspections will be completed by August 2016 for both Salem Units. Approximately one third of the 57 inspections will be	A.2.1.28	Program to be enhanced prior to the period of extended operation. Inspection Schedule identified in Commitment.	LRA Section B.2.1.28 Salem Letter LR-N10-0165 RAI B.2.1.28-1 RAI B.2.1.28-2 May 13, 2010 Salem Letter LR-N10-0244 RAI 3.5.2.2.1.7-01 July 15, 2010 Salem Letter LR-N10-0321 RAI B.2.1.28-04 RAI B.2.1.33-06 September 1, 2010 Salem Letter LR-N10-0382 October 15, 2010

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Item Number	**Commitment**	**Updated Final Safety Analysis Report (UFSAR) Supplement Section/ LRA Section**	**Enhancement or Implementation Schedule**	**Source**
	completed during each refuel outage (Salem Unit 1 involves the following refuel outages: Spring 2013, Fall 2014, and Spring 2016. Salem Unit 2 involves the following refuel outages: Fall 2012, Spring 2014, and Fall 2015). It is acceptable to perform greater than one third of the inspections in any refuel outage to accelerate the inspection schedule. During the period of extended operation • One containment liner insulation panel will be selected, at random, for removal from each quadrant, during each of the three Periods in an Inspection Interval. Therefore, a total of 12 containment liner insulation panels will be selected, in each unit, during each ten year Inspection Interval, to allow for examination of the containment liner behind the containment liner insulation. • The randomly selected containment liner insulation panels in each quadrant will not include containment liner insulation panels previously selected. 2. Visual inspection of 100% of the moisture barrier, at the junction between the containment concrete floor and the containment liner, will be performed in accordance with ASME Section XI, Subsection IWE program requirements, to the extent practical within the limitation of design, geometry, and materials of construction of the components. The bottom edge of the stainless steel insulation lagging will be trimmed, if necessary, to perform the moisture barrier inspections. This inspection will be performed prior to the period of extended operation, and on a frequency consistent with IWE inspection requirements thereafter. Should unacceptable degradation be found, corrective actions, including extent of condition, will be addressed in accordance with the corrective action process.			

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	As a follow-up to inspections performed during the 2009 refueling outage, the following specific corrective actions will be performed on Unit 2 prior to entry into the period of extended operation: • Examine the accessible ¾” knuckle plate. If corrosion is observed to extend below the surface of the moisture barrier, excavate the moisture barrier to sound metal below the floor level and perform examinations as required by IWE. • Perform remote visual inspections, of the six capped vertical leak chase channels, below the containment floor to determine extent of condition. • Remove the concrete floor and expose the ¼” containment liner plate (floor) for a minimum of two of the vertical leak chase channels with holes. Perform examination of exposed ¼” containment liner plate (floor) as required by IWE. Additional excavations will be performed, if necessary, depending upon conditions found at the first two channels. • Remove ½” containment liner insulation panels, adjacent to accessible areas where there are indications of corrosion, to determine the extent of condition of the existing corroded areas of the containment liner plate. • Perform augmented examinations of the areas of the ½” containment liner plate behind insulation panels, where loss of material was previously identified, in accordance with IWE-2420. • Examine 100% of the moisture barrier in accordance with IWE-2310 and replace or repair the moisture barrier to meet the acceptance standard in IWE-3510. As a follow-up to inspections performed during the 2010 refueling			

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	outage, the following specific corrective actions will be performed on Unit 1 prior to entry into the period of extended operation: • Perform augmented examinations of the ¾” containment liner (knuckle plate) at 78’ elevation in accordance with IWE-2420. • Perform augmented examinations of the areas of the ½” containment liner plate behind insulation panels, where loss of material was previously identified, in accordance with IWE-2420. • Remove ½” containment liner insulation panels, adjacent to accessible areas where there are indications of corrosion, to determine the extent of condition of the existing corroded areas of the containment liner plate. 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. 4. Owner augmented inspections will be performed at the Salem Unit 1 and Unit 2 area of the Containment liner, under the fuel transfer canal and behind the Containment liner insulation, which are subjected to leaks from the reactor cavity. These owner augmented inspections will be performed on a frequency of once per Containment Inservice Inspection Period, starting with the current Period. These owner augmented inspections will continue, under the IWE program, as long as leakage from the reactor cavity or fuel transfer canal is observed between the Containment liner and the Containment liner insulation, including during the PEO.			

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29	ASME Section XI, Subsection IWL, is an existing program that will be enhanced to include: 1. Examination and acceptance criteria in accordance with the guidance contained in ACI 349.3R.	A.2.1.29	Program to be enhanced prior to the period of extended operation.	LRA Section B.2.1.29 Salem Letter LR-N10-0165 RAI B.2.1.29-1 May 13, 2010
30	ASME Section XI, Subsection IWF	A.2.1.30	Ongoing	LRA Section B.2.1.30
31	10 CFR Part 50, Appendix J	A.2.1.31	Ongoing	LRA Section B.2.1.31
32	Masonry Wall is an existing program that will be enhanced to include: 1. Additional buildings and masonry walls as described in A.2.1.32. 2. Add an Examination Checklist for masonry wall inspection requirements. 3. Specify an inspection frequency of not greater than 5 years for masonry walls.	A.2.1.32	Program to be enhanced prior to the period of extended operation.	LRA Section B.2.1.32
33	Structures Monitoring is an existing program that will be enhanced to include: 1. Additional structures and components as described in A.2.1.33. 2. Concrete structures will be observed for a reduction in equipment anchor capacity due to local concrete degradation. This will be accomplished by visual inspection of concrete surfaces around anchors for cracking and spalling. 3. Clarify that inspections are performed for loss of material due to corrosion and pitting of additional steel components, such as embedments, panels and enclosures, doors, siding, metal deck, and anchors. 4. Require inspection of penetration seals, structural seals, and elastomers, for degradations that will lead to a loss of sealing by visual inspection of the seal for hardening, shrinkage and loss of	A.2.1.33	Program to be enhanced prior to the period of extended operation. Core sample Inspection schedule identified in commitment.	LRA Section B.2.1.33 Salem Letter LR-N10-0165 RAI B.2.1.33-1 RAI B.2.1.33-2 May 13, 2010 Salem Letter LR-N10-0321 RAI B.2.1.33-05 September 1, 2010 Salem Letter

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	strength. 5. Require the following actions related to the spent fuel pool liner: a. Perform periodic structural examination of the Fuel Handling Building per ACI 349.3R to ensure structural condition is in agreement with the analysis. b. Monitor telltale leakage and inspect the leak chase system to ensure no blockage. c. Test water drained from the telltales and seismic gap for boron, chloride, iron, and sulfate concentrations; and pH. Acceptance criteria will assess any degradation from the borated water. Sample readings outside the acceptance criteria will be entered into and evaluated in the corrective action program. d. Perform one shallow core sample in each of the Unit 1 Spent Fuel Pool walls (east and west) that have shown ingress of borated water through the concrete. The core samples will be examined for degradation from borated water. Also the core samples (east and west walls) will expose rebar, which will be examined for signs of corrosion. The core sample from the west wall will be taken by the end of 2013 and the core sample from the east wall will be taken by the end of 2015. e. Perform a structural examination per ACI 349.3R every 18 months of the Unit 1 Spent Fuel Pool wall in the sump room where previous inspections have shown ingress of borated water through the concrete. 6. Require monitoring of vibration isolators, associated with component supports other than those covered by ASME XI, Subsection IWF. 7. Add an Examination Checklist for masonry wall inspection requirements. 8. Parameters monitored for wooden components will be enhanced to include: Change in Material Properties, Loss of Material due to Insect Damage and Moisture Damage.			LRN-N10-0414 RAI B.2.1.33-07 December 14, 2010 Salem letter LR-N11-0041 RAI B.2.1.33-07 Update February 25, 2011

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	9. Specify an inspection frequency of not greater than 5 years for structures including submerged portions of the service water intake structure. 10. Require individuals responsible for inspections and assessments for structures to have a B.S. Engineering degree and/or Professional Engineer license, and a minimum of four years experience working on building structures. 11. Perform periodic sampling, testing, and analysis of ground water chemistry for pH, chlorides, and sulfates on a frequency of 5 years. Groundwater samples in the areas adjacent to Unit 1 containment structure and Unit 1 auxiliary building will also be tested for boron concentration. 12. Require supplemental inspections of the affected in scope structures within 30 days following extreme environmental or natural phenomena (large floods, significant earthquakes, hurricanes, and tornadoes). 13. Perform a chemical analysis of ground or surface water in-leakage when there is significant in-leakage or there is reason to believe that the in-leakage may be damaging concrete elements or reinforcing steel. 14. Implementing procedures will be enhanced to include additional acceptance criteria details specified in ACI 349.3R-96. 15. When the reactor cavity is flooded up, Salem will periodically monitor the telltales associated with the reactor cavity and refueling canal for leakage. If telltale leakage is observed, then the pH of the leakage will be measured to ensure that concrete reinforcement steel is not experiencing a corrosive environment. In addition, Salem will periodically inspect the leak chase system associated with the reactor cavity and refueling canal to ensure the telltales are free of significant blockage. Salem will also inspect concrete surfaces for degradation where leakage has been observed, in accordance with this Program.			

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34	RG 1.127, Inspection of Water-Control Structures Associated With Nuclear Power Plants is an existing program that will be enhanced to include: 1. Parameters monitored for wooden components will be enhanced to include change in material properties and loss of material due to insect damage and moisture damage. 2. Parameters monitored for elastomers will be enhanced to include hardening, shrinkage and loss of strength due to weathering and elastomer degradation. 3. The inspection requirement for submerged concrete structural components will be enhanced to require that inspections be performed by dewatering a pump bay or by a diver if the pump bay is not dewatered. 4. Specify an inspection frequency of not greater than 5 years for structures including submerged portions of the Service Water Intake Structure. 5. Require supplemental inspections of the in scope structures within 30 days following extreme environmental or natural phenomena (large floods, significant earthquakes, hurricanes, and tornadoes).	A.2.1.34	Program to be enhanced prior to the period of extended operation.	LRA Section B.2.1.34
35	Protective Coating Monitoring and Maintenance Program	A.2.1.35	Ongoing	LRA Section B.2.1.35

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36	Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements is a new program and will be used to manage aging of non-EQ cables and connections during the period of extended operation.	A.2.1.36	Program and initial inspections to be implemented prior to the period of extended operation.	LRA Section B.2.1.36
37	Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits is a new program that will be implemented to manage the aging of the cable and connection insulation of the in scope portions of the Radiation Monitoring System and the Reactor Protection System (i.e., the nuclear instrumentation system).	A.2.1.37	Program and initial assessment of testing and calibration results to be implemented prior to the period of extended operation.	LRA Section B.2.1.37
38	Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements is a new program that will be used to manage the aging effects and mechanisms of non-EQ, in scope inaccessible medium voltage cables (4,160V, 13,800V). The cable test frequency will be established based on test results and industry operating experience. The maximum time between tests will be no longer than 6 years. Manholes and cable vaults associated with the cables included in this aging management program will be inspected for water collection (with water removal as necessary) with the objective of minimizing the exposure of medium voltage cables to significant moisture. Prior to the period of extended operation, the frequency of inspections for accumulated water will be established based on inspection results to minimize the exposure of medium voltage cables to significant moisture. The maximum time between inspections will be no longer than one year.	A.2.1.38	Enhanced program, initial cable tests, and initial manhole and cable vault inspections to be implemented prior to the period of extended operation. Test and inspection schedule identified in commitment.	LRA Section B.2.1.38 Salem Letter LR-N10-0225 RAI B.2.1.38-01 July 8, 2010 Salem Letter LR-N10-0348 LRA Supplement October 7, 2010

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	The Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements aging management program will be enhanced as follows: 1. Change cable testing maximum frequency from 10 years to 6 years. Change manhole and cable vault inspection maximum frequency from 2 years to 1 year.			
39	Metal Enclosed Bus is a new program that will manage the aging of in-scope metal enclosed busses.	A.2.1.39	Program and initial inspections to be implemented prior to the period of extended operation.	LRA Section B.2.1.39
40	Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements is a new program that will be used to confirm the slow progression or the absence of an aging effect with respect to electrical cable connection stressors. A representative sample of non-EQ electrical cable connections will be selected, for one-time testing considering application (medium and low voltage), circuit loading (high loading) and location, with respect to connection stressors.	A.2.1.40	Program and one-time testing to be implemented prior to the period of extended operation.	LRA Section B.2.1.40
41	High Voltage Insulators is a new program that manages the degradation of insulator quality due to the presence of salt deposits or surface contamination.	A.2.2.1	Program to be implemented prior to the period of extended operation.	LRA Section B.2.2.1
42	Periodic Inspection is a new program that manages the aging of piping, piping components, piping elements, ducting components, tanks and heat exchanger components.	A.2.2.2	Program to be implemented prior to the period of extended operation.	LRA Section B.2.2.2
43	Aboveground Non-Steel Tanks is a new program that will manage loss of material of outdoor non-steel tanks. The Aboveground Non-Steel Tanks program will include a UT wall thickness inspection of the bottom of the tanks.	A.2.2.3	Program to be implemented prior to the period of	LRA Section B.2.2.3

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	The UT measurements will be taken to ensure that significant degradation is not occurring and that the component intended function will be maintained during the extended period of operation.		extended operation. Tank bottom UT inspections will also be performed prior to the period of extended operation.	
44	Buried Non-Steel Piping Inspection is an existing program that will be enhanced to include: 1. At least one (1) opportunistic or focused excavation and inspection will be performed on buried reinforced concrete piping and components during each ten (10) year period, beginning ten (10) years prior to entry into the period of extended operation. 2. At least one (1) opportunistic or focused excavation and inspection will be performed on buried stainless steel penetration bellows between the Containment Structure and the Fuel Handling Building, including the penetration sleeves, during each ten (10) year period, beginning ten (10) years prior to entry into the period of extended operation. 3. Guidance for inspection of concrete aging effects.	A.2.2.4	Program to be enhanced prior to the period of extended operation. Inspection Schedule identified in Commitment	LRA Section B.2.2.4 Salem Letter LR-N10-0322 RAI B.2.1.22 September 7, 2010
45	Boral Monitoring is an existing program that will be enhanced to include: 1. The program will be enhanced to perform a neutron attenuation measurement on one each of the three (no vent holes, one vent holes and two vent holes) flat plate sandwich Boral test coupons during the first three two-year inspection frequency periods and every six years thereafter for the Exxon spent fuel storage rack assemblies. 2. The program will be enhanced to include acceptance criteria of the neutron attenuation measurement on the Boral test coupons for the Exxon spent fuel storage rack assemblies: A decrease of no more	A.2.2.5	Program to be enhanced prior to the period of extended operation. Inspection Schedule identified in Commitment.	LRA Section B.2.2.5

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	than 5% in Boron-10 content as determined by neutron attenuation measurements. The benchmark Boron-10 content used for comparison will be based on the nominal B-10 areal density in the design basis specification.			
46	Nickel Alloy Aging Management	A.2.2.6	Ongoing	LRA Section B.2.2.6
47	Metal Fatigue of the Reactor Coolant Pressure Boundary is an existing program that will be enhanced to include: 1. Adding transients beyond those defined in the Technical Specifications and the UFSAR, and expanding the fatigue monitoring program to encompass other components identified to have fatigue as an analyzed aging effect, which require monitoring. 2. Using a software program to automatically count transients and calculate cumulative usage on select components. 3. Addressing the effects of the reactor coolant environment on component fatigue life by assessing the impact of the reactor coolant environment on a sample of critical components for the plant identified in NUREG/CR-6260. 4. Requiring a review of additional reactor coolant pressure boundary locations if the usage factor for one of the environmental fatigue sample locations approaches its design limit.	A.3.1.1	Program to be enhanced prior to the period of extended operation.	LRA Section B.3.1.1
48	Environmental Qualification of Electric Components (EQ)	A.3.1.2	Ongoing	LRA Section B.3.1.2
49	Revised Pressure-Temperature (P-T) limits will be submitted to the NRC when necessary to comply with 10 CFR 50 Appendix G.	A.4.2.4	Ongoing	LRA Section 4.2.4
50	Steam Generator Divider Plate Inspection Salem will perform an inspection of each of the four (4) Unit 1 steam generators to assess the condition of the divider plate assembly. The examination technique(s) used will be capable of detecting primary water stress corrosion cracking (PWSCC) in the steam generator divider plate	Not Applicable	Prior to August 2026	Salem Letter LR-N10-0369 RAI 3.1.1-02 October 7, 2010

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	assemblies and the associated welds. The steam generator divider plate inspections will be completed within the first ten (10) years of the Salem Unit 1 period of extended operation.			
51	Steam Generator Tube-to-Tubesheet Weld Cracking Salem will develop a plan for each Unit to address the potential for cracking of the primary to secondary pressure boundary due to PWSCC of tube-to-tubesheet welds. Each plan will consist of two options: <u>Salem Unit 1</u> Option 1 (Analysis): Salem Unit 1 will obtain permanent approval for Alternate Repair Criteria from the NRC, **or** Option 2 (inspection): Salem Unit 1 will perform a One-Time inspection of a representative number of tube-to-tubesheet welds in each of the four (4) steam generators to determine if PWSCC is present. If weld cracking is identified, a) the condition will be resolved through repair or engineering evaluation to justify continued service, as appropriate, and b) a periodic monitoring program will be established to perform routine tube-to-tubesheet inspections for the remaining life of the steam generators. <u>Salem Unit 2</u> Option 1 (Analysis):	Not Applicable	Develop a plan prior to the Period of Extended Operation for each Unit. If the analysis option is chosen, implement the requirements of the plan, including obtaining any required NRC approval, by April 2018 for Unit 1, and by April 2028 for Unit 2. If steam generator inspections are to be performed, they will be performed between April 2018 and April 2023 for Unit 1, and April 2028 and April 2033 for Unit 2.	Salem Letter LR-N10-0421 RAI 3.1.1-03 December 1, 2010 Salem Letter LR-N10-0438 Revised Response to RAI 3.1.1-03 December 15, 2010

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	Salem Unit 2 will perform an analytical evaluation either determining that the tubesheet cladding and welds are not susceptible to PWSCC, or redefining the pressure boundary of the tubes, where the steam generator tube-to-tubesheet welds are not required for the reactor coolant pressure boundary function. The redefinition of the reactor coolant pressure boundary will be submitted as part of a license amendment request requiring approval from the NRC, **or** Option 2 (inspection): Salem Unit 2 will perform a One-Time inspection of a representative number of tube-to-tubesheet welds in each of the four (4) steam generators to determine if PWSCC is present. If weld cracking is identified, a) the condition will be resolved through repair or engineering evaluation to justify continued service, as appropriate, and b) a periodic monitoring program will be established to perform routine tube-to-tubesheet inspections for the remaining life of the steam generators.			
52	Salem will perform a review of design basis ASME Code Class 1 fatigue evaluations to determine whether the NUREG/CR-6260 based locations that have been evaluated for the effects of the reactor coolant environment on fatigue usage are the limiting locations for the Salem plant configuration. If more limiting locations are identified, the most limiting location will be evaluated for the effects of the reactor coolant environment on fatigue usage. If any of the limiting locations consist of nickel alloy, NUREG/CR-6909 methodology for nickel alloy will be used in the evaluation	Not Applicable	Prior to the period of extended operation.	Salem Letter LR-N10-0445 RAI 4.3-08 December 21, 2010
53	Salem Fatigue Calculations using WESTEMS™ program Salem will include written explanation and justification of any user intervention in future evaluations using the WESTEMS "Design CUF" (NB-3200 module).	A.4.3.7	Within 60 days of issuance of the renewed operating license	Salem Letter LR-N11-0042 January 31, 2011 Salem Letter LR-N11-0057

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				February 24, 2011
54	Salem Fatigue Calculations using WESTEMS™ program Salem will not use or implement the NB-3600 option (module) of the WESTEMS™ program in future online fatigue monitoring and design calculations.	A.4.3.7	Within 60 days of issuance of the renewed operating license	Salem Letter LR-N11-0042 January 31, 2011 Salem Letter LR-N11-0057 February 24, 2011