APPENDIX B

LICENSE RENEWAL FINAL SAFETY ANALYSIS REPORT SUPPLEMENT

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A.1 Introduction

The application for a renewed operating license is required by 10 CFR 54.21(d) to include a FSAR Supplement. This appendix, which includes the following sections, comprises the FSAR supplement:

- Section A.1.1 contains a listing of the aging management programs that correspond to NUREG-1801 Chapter XI programs.
- Section A.1.2 contains a listing of the plant-specific aging management programs.
- Section A.1.3 contains a listing of aging management programs that correspond to NUREG-1801 Chapter X programs associated with Time-Limited Aging Analyses.
- Section A.1.4 contains a listing of the Time-Limited Aging Analyses (TLAA).
- Section A.1.5 contains a discussion of the Quality Assurance Program and Administrative Controls.
- Section A.2 contains a summarized description of the aging management programs.
- Section A.2.1 contains a summarized description of the NUREG-1801 Chapter XI programs for managing the effects of aging.
- Section A.2.2 contains a summarized description of the plant-specific programs for managing the effects of aging.
- Section A.3 contains a summarized description of the NUREG-1801 Chapter X programs that support the TLAAs.
- Section A.4 contains a summarized description of the TLAAs applicable to the period of extended operation.
- Section A.5 contains the License Renewal Commitment List.

The integrated plant assessment for license renewal identified new and existing aging management programs necessary to provide reasonable assurance that systems, structures, and components within the scope of license renewal will continue to perform their intended functions consistent with the Current Licensing Basis (CLB) for the period of extended operation. The period of extended operation is defined as 20 years from the units' original operating license expiration date.

A.1.1 NUREG-1801 Chapter XI Aging Management Programs

The NUREG-1801 Chapter XI Aging Management Programs (AMPs) are described in the following sections. The AMPs are either consistent with generally accepted industry methods as discussed in NUREG-1801 or require enhancements.

Commitments for program additions and enhancements are identified in the appropriate sections.

1. ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (Section A.2.1.1)

2. Water Chemistry (Section A.2.1.2)

3. Reactor Head Closure Studs (Section A.2.1.3)

4. Boric Acid Corrosion (Section A.2.1.4)

5. Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors (Section A.2.1.5)

6. Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) (Section A.2.1.6)

7. PWR Vessel Internals (Section A.2.1.7)

8. Flow-Accelerated Corrosion (Section A.2.1.8)

9. Bolting Integrity (Section A.2.1.9)

10. Steam Generator Tube Integrity (Section A.2.1.10)

11. Open-Cycle Cooling Water System (Section A.2.1.11)

12. Closed-Cycle Cooling Water System (Section A.2.1.12)

13. Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems (Section A.2.1.13)

14. Compressed Air Monitoring (Section A.2.1.14)

15. Fire Protection (Section A.2.1.15)

16. Fire Water System (Section A.2.1.16)

17. Aboveground Steel Tanks (Section A.2.1.17)

18. Fuel Oil Chemistry (Section A.2.1.18)

19. Reactor Vessel Surveillance (Section A.2.1.19)

20. One-Time Inspection (Section A.2.1.20)

21. Selective Leaching of Materials (Section A.2.1.21)

22. Buried Piping Inspection (Section A.2.1.22)

23. One-Time Inspection of ASME Code Class 1 Small Bore-Piping (Section A.2.1.23)

24. External Surfaces Monitoring (Section A.2.1.24)

25. Flux Thimble Tube Inspection (Section A.2.1.25)
26. Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (Section A.2.1.26)
27. Lubricating Oil Analysis (Section A.2.1.27)
28. ASME Section XI, Subsection IWE (Section A.2.1.28)
29. ASME Section XI, Subsection IWL (Section A.2.1.29)
30. ASME Section XI, Subsection IWF (Section A.2.1.30)
31. 10 CFR Part 50, Appendix J (Section A.2.1.31)
32. Masonry Wall Program (Section A.2.1.32)
33. Structures Monitoring Program (Section A.2.1.33)
34. RG 1.127 Inspection of Water-Control Structures (Section A.2.1.34)
35. Protective Coating Monitoring and Maintenance Program (Section A.2.1.35)
36. Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (Section A.2.1.36)
37. Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits (Section A.2.1.37)
38. Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (Section A.2.1.38)
39. Metal Enclosed Bus (Section A.2.1.39)
40. Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (Section A.2.1.40)

A.1.2 Plant-Specific Aging Management Programs

The plant-specific programs are described in the following sections. Commitments for program additions and enhancements are identified in Section A.5 License Renewal Commitment List.

1. High Voltage Insulators (Section A.2.2.1)
2. Periodic Inspection (Section A.2.2.2)
3. Aboveground Non-Steel Tanks (Section A.2.2.3)

4. Buried Non-Steel Piping Inspection (Section A.2.2.4)
5. Boral Monitoring Program (Section A.2.2.5)
6. Nickel Alloy Aging Management Program (Section A.2.2.6)

A.1.3 NUREG-1801 Chapter X Aging Management Programs

The NUREG-1801 Chapter X Aging Management Programs associated with Time-Limited Aging Analyses are described in the following sections. The AMPs are either consistent with generally accepted industry methods as discussed in NUREG-1801 Chapter X or require enhancements. Commitments for program additions and enhancements are identified in the Section A.5 License Renewal Commitment List.

1. Metal Fatigue of Reactor Coolant Pressure Boundary (Section A.3.1.1)
2. Environmental Qualification (EQ) of Electric Components (Section A.3.1.2)

A.1.4 Time-Limited Aging Analyses

Summaries of the Time-Limited Aging Analyses applicable to the period of extended operation are included in the following sections.

1. Reactor Vessel Neutron Embrittlement (Section A.4.2)
2. Metal Fatigue of Piping and Components (Section A.4.3)
3. Other Plant-Specific Analyses (Section A.4.4)
4. Fuel Transfer Tube Bellows Design Cycles (Section A.4.5)
5. Crane Load Cycle Limits (Section A.4.6)
6. Environmental Qualification of Electric Equipment (Section A.4.7)

A.1.5 Quality Assurance Program and Administrative Controls

The Quality Assurance Program implements the requirements of 10 CFR 50, Appendix B, and is consistent with the summary in Appendix A.2, "Quality Assurance for Aging Management Programs (Branch Technical Position IQMB-1)" of NUREG-1800. The Quality Assurance Program includes the elements of corrective action, confirmation process, and administrative controls, and these elements are applicable to the safety-related and non-safety related systems, structures, and components (SSCs) that are subject to Aging Management Review (AMR). In many cases, existing activities were found adequate for managing aging effects during the period of extended operation.

A.2 Aging Management Programs

A.2.1 NUREG-1801 Chapter XI Aging Management Programs

This section provides summaries of the NUREG-1801 programs credited for managing the effects of aging.

A.2.1.1 ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD

The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD aging management program is an existing program that consists of periodic volumetric and visual examinations of components for assessment, identification of signs of degradation, and establishment of corrective actions. The program includes inspections performed to manage cracking, loss of fracture toughness and loss of material in Class 1, 2, and 3 piping and components exposed to reactor coolant, steam, treated water and treated borated water environments. The inspections will be implemented in accordance with 10 CFR 50.55(a). These activities include inspections, and monitoring and trending of results to confirm that aging effects are managed.

A.2.1.2 Water Chemistry

The Salem Water Chemistry aging management program is an existing program that provides activities for monitoring and controlling the chemical environments of the Salem primary cycle and secondary cycle systems such that aging effects of system components are minimized. Aging effects include cracking, loss of material, reduction of neutron-absorbing capacity and reduction of heat transfer. The primary cycle scope of this program consists of the reactor coolant system and related auxiliary systems containing treated water, reactor coolant, treated borated water and steam, including the primary side of the steam generators that contain treated water and steam. The secondary cycle portion of the program consists of the various secondary side systems and the secondary side of the steam generators. Major component types include reactor vessel, reactor internals, piping, piping elements and piping components, heat exchangers and tanks. The Water Chemistry aging management program is consistent with EPRI, Pressurized Water Reactor Primary Chemistry Guidelines, and Plant UFSAR limits for fluorides, chlorides, and dissolved oxygen. The Water Chemistry program is consistent with EPRI, Pressurized Water Secondary Water Chemistry Guidelines.

The One-Time Inspection program will be used to verify the effectiveness of the Water Chemistry program in managing the aging effects for stainless steel components in a treated borated water environment where dissolved oxygen may not be controlled to less than 100 ppb.

A.2.1.3 Reactor Head Closure Studs

The Reactor Head Closure Studs program is an existing program that provides for condition monitoring and preventive activities to manage reactor head closure stud cracking caused by stress corrosion cracking. The program is implemented through station procedures based on the examination and inspection requirements specified in ASME Section XI, Table IWB-2500-1 and preventive measures described in NRC Regulatory Guide 1.65, "Materials and Inspection for Reactor Vessel Closure Studs."

A.2.1.4 Boric Acid Corrosion

The Boric Acid Corrosion aging management program is an existing program that manages loss of; material on piping, piping elements, heat exchangers, bolting, panels, racks, cabinets and enclosures, insulation jacketing, cable trays, concrete anchors, concrete embedments, manhole covers, conduit, miscellaneous steel, and other structural components, component supports, cracking, blistering, flaking, peeling, and delamination of coatings, and corrosion of electrical connector contact surfaces. The program includes provisions to identify, inspect, examine and evaluate leakage, and initiate corrective action. The program relies in part on implementation of recommendations contained in NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Components in PWR plants".

A.2.1.5 Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors

The Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors aging management program is an existing program that consists of a combination of periodic bare metal visual inspections of the outer surface of the upper reactor vessel closure head (closure head) and periodic non-destructive examinations (surface and volumetric) of the upper vessel head penetration (VHP) nozzles and associated J-groove welds, for assessment, identification of signs of degradation, and establishment of corrective actions. The program monitors the condition of nickel-alloy components and J-groove welds for the effects of cracking in a reactor coolant environment. The inspections will be implemented in accordance with 10 CFR 50.55(a), which endorses ASME Code Case N-729-1. The program ensures the structural integrity of the closure head, VHP nozzles, and associated J-groove welds, and the detection of cracking and any loss of material/wastage prior to a loss of intended function. These activities include examinations, and monitoring and trending of results to confirm that aging effects are managed.

A.2.1.6 Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)

The Thermal Aging Embrittlement of Cast Austenitic Stainless Steel aging management program is a new program that includes condition monitoring activities to provide assurance that reactor coolant system CASS components susceptible to thermal aging embrittlement meet the intended functions. The reactor coolant system CASS components are maintained by inspecting and

evaluating the extent of thermal aging embrittlement in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, 1998 Edition, 2000 Addenda. The Salem ASME Section XI Inservice Inspection program is augmented by the implementation of the Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) aging management program which monitors the aging effect of loss of fracture toughness due to thermal aging embrittlement of CASS components.

The Thermal Aging Embrittlement of CASS 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 management will be accomplished through either a component-specific flaw tolerance evaluation or enhanced volumetric examinations. Inspections or evaluations are not required for components that are determined not to be susceptible to thermal aging embrittlement. Screening for CASS components susceptible to thermal aging embrittlement is not required for pump casings and valve bodies. The existing ASME Section XI inspection requirements are adequate for managing the aging effects of Class 1 pump casings and valve bodies. This new program will be implemented prior to the period of extended operation.

A.2.1.7 PWR Vessel Internals

Salem Units 1 and 2 commit to the following activities for the new PWR Vessel Internals program:

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.

This new aging management program will be implemented prior to the period of extended operation.

A.2.1.8 Flow-Accelerated Corrosion

The Flow-Accelerated Corrosion (FAC) aging management program at Salem is an existing program based on EPRI guidelines in NSAC-202L, "Recommendations for an Effective Flow Accelerated Corrosion Program." The program provides for predicting, detecting, and monitoring wall thinning in piping, fittings, valve bodies, and heat exchangers due to FAC. Analytical evaluations and periodic examinations of locations that are most susceptible to wall thinning due to FAC are used to predict the amount of wall thinning in pipes, fittings, and feedwater heater shells. Program activities include analyses to determine critical locations, baseline inspections to determine the extent of

thinning at these critical locations, and follow-up inspections to confirm the predictions. Repairs and replacements are performed as necessary.

A.2.1.9 Bolting Integrity

The Bolting Integrity aging management program is an existing program that provides for aging management of pressure retaining bolted joints, component support bolting and structural bolting within the scope of license renewal. The Bolting Integrity program incorporates NRC and industry recommendations delineated in NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants," EPRI TR-104213, "Bolted Joint Maintenance & Applications Guide," and EPRI NP 5769, "Degradation and Failure of Bolting in Nuclear Power Plants," as part of the comprehensive corporate component bolting program. The program provides for managing cracking, loss of material and loss of preload of bolted joints in the following environments: air, groundwater/soil, raw water, treated borated water, and soil. Included in the aging management activities directed by this program are visual inspections for pressure retaining bolted joint leakage and preventive measures.

The Bolting Integrity aging management program 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.

This enhancement will be implemented prior to the period of extended operation.

A.2.1.10 Steam Generator Tube Integrity

The Steam Generator Tube Integrity aging management program is an existing program that establishes the operation, maintenance, testing, inspection and repair of the steam generators to ensure that Technical Specification surveillance requirements, ASME Code requirements and the Maintenance Rule performance criteria are met, thereby adequately managing the aging effects of the steam generator tubes, plugs, and tube support plates. Aging effects include cracking, loss of material, loss of preload, reduction of heat transfer and wall thinning. The program identifies and maintains the steam generator design and licensing bases and implements NEI 97-06. NEI 97-06 establishes a framework for prevention, inspection, evaluation, repair and leakage monitoring measures.

A.2.1.11 Open-Cycle Cooling Water System

The Salem Open-Cycle Cooling Water System aging management program is an existing program that includes mitigative, performance-monitoring, and condition-monitoring activities to manage the internal corrosion of piping to minimize susceptibility of corrosion and to verify that corrosion has not exceeded acceptance limits. More than one type of aging management program is necessary to ultimately ensure that the aging effects are adequately managed and the intended function(s) are maintained for the extended period of operation. These activities provide assurance that cracking, material loss, and heat transfer reduction aging effects are maintained at acceptable levels for systems and components within the scope of license renewal. The GL 89-13 activities provide for management of aging effects in raw water cooling systems through tests and inspections per the guidelines of NRC Generic Letter 89-13. System and component testing, visual inspections, non-destructive examination (e.g., RT-Radiographic Testing, UT-Ultrasonic Testing, and/or ECT-Eddy Current Testing), and sodium hypochlorite injection are conducted to ensure that aging effects are managed such that system and component intended functions and integrity are maintained. Major component types include pumps, piping, piping elements, piping components, heat exchangers and tanks.

The Salem Open-Cycle Cooling Water System (OCCWS) aging management program primarily consists of station GL 89-13 activities that include sodium hypochlorite injection, system testing, periodic inspections and non-destructive examination. The program includes surveillance and control techniques to manage aging effects caused by bio-fouling, corrosion, erosion, protective coating failures, and silting in the Service Water System components and on the systems, structures, and components supported by the Service Water System. Other activities include station maintenance inspections, component preventive maintenance, plant surveillance testing, and inspections. These activities provide for management of loss of material (without credit for protective coatings) and heat transfer reduction (including fouling from biological, corrosion product, and external sources) aging effects where applicable in system components exposed to a raw water environment.

A.2.1.12 Closed-Cycle Cooling Water System

The Closed-Cycle Cooling Water System aging management program is an existing program that manages aging of piping, piping components, piping elements, tanks, and heat exchangers that are included in the scope of license renewal for loss of material, stress corrosion cracking, and reduction of heat transfer and are exposed to a closed cooling water environment at Salem. The Closed-Cycle Cooling Water System aging management program relies on mitigation measures to minimize corrosion by maintaining inhibitors and by performing non-chemistry monitoring consisting of inspection and nondestructive examinations based on industry-recognized guidelines of EPRI 1007820 for closed-cycle cooling water systems. Station maintenance inspections and nondestructive examinations provide condition monitoring of heat exchangers exposed to closed-cycle cooling water environments.

The following enhancements will be incorporated to the Closed-Cycle Cooling Water System Program.

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 monitoring 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 the Component Cooling System.

5. New recurring tasks will be established for enhancing the 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.

These enhancements will be implemented prior to entering the period of extended operation. In addition, the one-time inspections will be performed prior to the period of extended operation.

A.2.1.13 Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems

The Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems aging management program is an existing program that is credited for managing aging effects of cranes and hoists in the scope of license renewal. Administrative controls ensure that only allowable loads are handled. Cranes and hoists structural components, including the bridge, the trolley, bolting, lifting devices, and the rail system are visually inspected periodically for loss of material. Bolting is also monitored for loss of preload by inspecting for missing, detached, or loosened bolts. The program relies on procurement controls and installation practices, defined in plant procedures, to ensure that only approved lubricants and proper torque are applied to bolting.

The Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems Program will be enhanced to include:

1. The program will be enhanced to include 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. The program will be enhanced to require visual inspection of the rails in the rail system for loss of material due to wear.
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.

These enhancements will be implemented prior to the period of extended operation.

A.2.1.14 Compressed Air Monitoring

The Compressed Air Monitoring aging management program is an existing program that manages piping, piping components, and piping elements, compressor housings, and tanks for loss of material due to general, pitting, and crevice corrosion in the Compressed Air System. The Compressed Air Monitoring aging management activities consist of preventive maintenance and condition monitoring measures to manage the aging effects.

A.2.1.15 Fire Protection

The Fire Protection aging management program is an existing program that includes a fire barrier inspection, diesel-driven fire pump inspection and Halon and Carbon Dioxide systems inspections and functional tests. These inspections and functional tests provide assurance that the fire protection components within the scope of license renewal are maintained operational. The fire protection components are comprised of piping, piping elements, piping components, doors, dampers and fire barriers. The Fire Protection program provides for visual inspections of fire barriers and penetration seals for signs of degradation such as cracking, loss of material and hardening, through periodic inspection and functional testing. These components within the scope of license renewal are maintained in accordance to the guidance contained within NFPA Codes and Standards. The fire barrier inspections require periodic visual inspections of fire barrier penetration seals, fire barrier walls, ceilings, and floors, and periodic visual inspection. Functional testing and inspections of the fire rated doors and dampers is performed to ensure that their operability is maintained. The program includes surveillance tests of fuel oil systems for the diesel-driven fire pumps to ensure that the fuel supply lines can perform their intended functions. The program also includes visual inspections and periodic operability tests of Halon and Carbon Dioxide fire suppression systems using NFPA Codes and Standards for guidance.

The Fire Protection aging management program will be enhanced to include:

1. The Salem 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 Salem 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.

These enhancements will be implemented prior to the period of extended operation.

A.2.1.16 Fire Water System

The Fire Water System aging management program is an existing program that provides for system pressure monitoring, fire system header flushing and flow testing, pump performance testing, hydrant flushing, and visual inspection activities. Testing, inspection and flushing activates are performed periodically assure that the aging effect of loss of material due to corrosion, microbiologically influenced corrosion (MIC), or biofouling are managed such that the system intended functions are maintained. System flow tests measure hydraulic resistance and compare results with previous testing, as a means of evaluating the internal piping conditions. Major component types include piping and fittings, heat exchangers, tanks and pumps. Monitoring system piping flow characteristics ensures that signs of loss of material will be detected in a timely manner. Pump performance tests, hydrant flushing and system inspections are based on guidance from the applicable NFPA standards.

The Fire Water System program will be enhanced as follows:

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.

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.

These enhancements will be implemented prior to the period of extended operation, with the inspections and testing performed in accordance with the schedule described above.

A.2.1.17 Aboveground Steel Tanks

The Aboveground Steel Tanks aging management program is an existing program that will manage loss of material aging effects of outdoor carbon steel tanks (Fire Protection Water Storage Tanks). Paint is a corrosion preventive measure, and periodic visual inspections will monitor degradation of the paint and any resulting metal degradation of carbon steel tanks. Inspection of the grout or sealant at the tank-foundation interface is included for tanks that are located on a sand bed on top of concrete foundations.

The Aboveground Steel Tanks program will be enhanced as follows:

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.

These enhancements will be implemented prior to the period of extended operation. Tank bottom UT inspections will also be performed prior to the period of extended operation.

A.2.1.18 Fuel Oil Chemistry

The Fuel Oil Chemistry aging management program is an existing program that includes preventive activities to provide assurance that contaminants are maintained at acceptable levels in fuel oil for systems and components within the scope of License Renewal. The fuel oil tanks within the scope of License Renewal are maintained by monitoring and controlling fuel oil contaminants in accordance with the guidelines of the American Society for Testing and Materials (ASTM). Fuel oil sampling and analysis is performed in accordance with approved procedures for new fuel oil and stored fuel oil. Fuel oil tanks are periodically drained of accumulated water and sediment, cleaned, and internally inspected. These activities effectively manage the effects of aging by providing reasonable assurance that potentially harmful contaminants are maintained at low concentrations.

The Fuel Oil Chemistry aging management program 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 inhibitors as determined by fuel oil sampling or inspection activities.

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.

These enhancements will be implemented prior to the period of extended operation. In addition, the one-time inspections will be performed prior to the period of extended operation.

A.2.1.19 Reactor Vessel Surveillance

The Reactor Vessel Surveillance Program is an existing program that manages the loss of fracture toughness due to neutron irradiation embrittlement of the reactor vessel beltline materials. The program fulfills the intent and scope of 10 CFR 50, Appendix H. This program evaluates neutron embrittlement by projecting Upper Shelf Energy (USE) for all reactor materials with projected neutron exposure greater than 10^{17} n/cm2 (E > 1.0 MeV) after 60 years of operation and with the development of pressure-temperature limit curves. Embrittlement information is obtained in accordance with Regulatory Guide 1.99, Rev. 2. In accordance with 10 CFR Part 50, Appendix H, Salem Units 1 and 2 will submit their proposed capsule withdrawal schedules for approval prior to implementation.

The Reactor Vessel Surveillance program will be enhanced as follows:

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 x 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, 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.): 1.00×10^{20} n/cm^2 (E > 1.0 MeV)

These enhancements will be implemented prior to the period of extended operation.

A.2.1.20 One-Time Inspection

The One-Time Inspection aging management program is a new program that will provide reasonable assurance that an aging effect is not occurring, or that the aging effect is occurring slowly enough to not affect a component intended function during the period of extended operation, and therefore will not require additional aging management. The program will be credited for cases where either (a) an aging effect is not expected to occur but there is insufficient data to completely rule it out, (b) an aging effect is expected to progress very slowly in the specified environment, but the local environment may be more adverse than that generally expected, or (c) the characteristics of the aging effect include a long incubation period. Major component types covered by the program include piping, piping elements and piping components, steam generators, heat exchangers and tanks.

The One-Time Inspection aging management program 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, 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, 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. Inspection methods will include visual examination or volumetric examinations. Acceptance criteria are in accordance with industry guidelines, codes, and standards, including the applicable edition of ASME Boiler and Pressure Vessel Code, Section XI. The One-Time Inspection program provides for the evaluation of the need for follow-up examinations to monitor the progression of aging if age-related degradation is found that could jeopardize an intended function before the end of the period of extended operation. Should aging effects be detected, the program triggers actions to characterize the nature and extent of the aging effect and determines what subsequent monitoring is needed to ensure intended functions are maintained during the period of extended operation.

The new program, including performance of physical inspections and evaluation of results, will be implemented prior to the period of extended operation.

A.2.1.21 Selective Leaching of Materials

The Selective Leaching of Materials aging program 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 in susceptible material and environment combinations. The program will also include aging management activities, for material and environment combinations where selective leaching is identified, to manage loss of material due to selective leaching. Components include valve bodies, filter housing, heat exchanger components, pump casings, strainer bodies, piping and fittings, drain traps, and tanks. One-time inspections will include visual examinations, supplemented by hardness tests, and other examinations, as required. If selective leaching is found, the condition will be evaluated to determine the need to expand inspection scope.

One-time inspections of susceptible material and environment combinations, where selective leaching has not previously been confirmed, will be performed in the last 10 years of the current term, prior to entering the period of extended operation. 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. For material and environment combinations where selective leaching is identified, aging management activities, such as periodic inspections, will be implemented to manage aging such that the component intended function is maintained consistent with the current licensing basis through the period of extended operation.

A.2.1.22 Buried Piping Inspection

The Buried Piping Inspection aging management program is an existing program that manages the external surface aging effects of loss of material for piping and components in a soil or groundwater (external) environment. The Salem buried component activities consist of preventive and condition-monitoring measures to manage, detect and monitor the loss of material due to external corrosion for piping and components in the scope of license renewal that are in a soil environment.

External inspections of buried components will occur opportunistically when they are excavated during maintenance.

The Buried Piping Inspection aging management program will be enhanced to include:

1. A study will be performed prior to entering the period of extended operation to assess the possibility and benefits of installing a cathodic protection 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 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 for 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 inspection 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 one (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 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.

These enhancements will be implemented prior to the period of extended operation, with the inspections performed in accordance with the schedule described above.

A.2.1.23 One-Time Inspection of ASME Code Class 1 Small Bore-Piping

The One-Time Inspection of ASME Code Class 1 Small-Bore Piping program 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 components include piping, piping elements and piping components. This program is a part of the Salem Risk Informed Inservice Inspection program.

The program will manage the aging effect through the identification and evaluation of cracking in small-bore Class 1 piping. The program will include one-time volumetric examination of a sample of class 1 butt welds for pipe size less than 4 inches NPS and greater than or equal to NPS 1. In addition, 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

- Two Class 1 small bore socket welds (one per unit) for cracking caused by thermal fatigue (thermal and mechanical loading)

The current examination method and frequency, VT-2, as allowed per the code, each refueling outage as defined in Code Case N-578-1, Table 1 will continue to be conducted. Any cracking identified in small-bore Class 1 piping resulting from stress corrosion or thermal and mechanical loading will result in periodic inspections. The program will effectively manage the aging effect by identifying and evaluating cracking in small-bore Class 1 piping prior to loss of intended function.

The One-Time Inspection of ASME Code Class 1 Small-Bore Piping Program will be implemented and one-time inspections completed and evaluated prior to the period of extended operation.

A.2.1.24 External Surfaces Monitoring

The External Surfaces Monitoring aging management program is a new program that directs visual inspections that are performed during system walkdowns. The program consists of periodic visual inspection of components such as piping, piping components, ducting, and other components within the scope of license renewal. The program manages aging effects through visual inspection of external surfaces for evidence of loss of material. Loss of material due to boric acid corrosion is managed by the Boric Acid Corrosion program. The external surfaces of components that are buried are inspected via the Buried Piping Inspection and Buried Non-Steel Piping Inspection programs. The external surfaces of above ground tanks are inspected via the Aboveground Steel Tanks and Aboveground Non-Steel Tanks programs.

This new aging management program will be implemented prior to the period of extended operation.

A.2.1.25 Flux Thimble Tube Inspection

The Flux Thimble Tube Inspection Program is a new program that manages the loss of material of the flux thimble tube materials by use of inspection methods such as eddy current testing. The reason it is a new program is that in Salem's response and supplements to NRC Bulletin 88-09, Salem implemented a flux thimble tube inspection program in 1988, then discontinued the program in 1993 after replacement of the flux thimble tubes with those of an improved design and satisfactory follow-up inspections of the improved materials for Unit 1. This new program implements the recommendations of NRC Bulletin 88-09, "Thimble Tube Thinning in Westinghouse Reactors". Acceptance criteria were established from industry guidance.

This new aging management program will be implemented prior to the period of extended operation.

A.2.1.26 Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components

The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components aging management program is a new program that manages the aging of the internal surfaces of steel piping, piping components and piping elements, ducting components, tanks and heat exchanger components. This

program will manage the aging effect of loss of material. The program includes provisions for visual inspections of the internal surfaces of components not managed under other aging management programs. Identified deficiencies due to age related degradation are evaluated under the Corrective Action Program.

This new aging management program will be implemented prior to the period of extended operation.

A.2.1.27 Lubricating Oil Analysis

The Lubricating Oil Analysis aging management program is an existing program that provides oil condition monitoring activities to manage the loss of material and the reduction of heat transfer in piping, piping components, piping elements, heat exchangers, and tanks within the scope of license renewal exposed to a lubricating oil environment. Sampling, analysis, and condition monitoring activities identify specific wear products and contamination and determine the physical properties of lubricating oil within operating machinery. These activities are used to verify that the wear product and contamination levels and the physical properties of the lubricating oil are maintained within acceptable limits to ensure that intended functions are maintained.

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.

Examination methods include visual and volumetric testing as required by ASME Section XI, Subsection IWE. Observed conditions that have the potential for impacting an intended function are evaluated for acceptability in accordance with ASME requirements or corrected in accordance with corrective action process.

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

1. Inspection of a sample of the inaccessible liner covered by insulation and lagging 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 completed during each refueling 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.

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 3/4" 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 1/4" containment liner plate (floor) for a minimum of two of the vertical leak chase channels with holes. Perform examination of exposed 1/4" 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 1/2" 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 area of the 1/2" 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 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 3/4" containment liner (knuckle plate) at 78' elevation in accordance with IWE-2420.

- Perform augmented examinations of the areas of the 1/2" containment liner plate behind insulation panels, where loss of material was previously identified, in accordance with IWE-2420.

- Remove 1/2" 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.

These enhancements will be implemented prior to the period of extended operation, with the inspections performed in accordance with the schedule described above.

A.2.1.29 ASME Section XI, Subsection IWL

The ASME Section XI, Subsection IWL aging management program is an existing program based on ASME Code and complies with the provisions of 10 CFR 50.55a. The program requires periodic inspection of Containment Structure concrete surfaces to identify areas of deterioration and distress such as defined in ACI 201.1, including loss of material, cracks and distortion, and loss of bond.

Inspection methods, inspected parameters, and acceptance criteria are in accordance with ASME Section XI, Subsection IWL as approved by 10 CFR 50.55a. Observed conditions that have the potential for impacting an intended function are evaluated for acceptability in accordance with ASME Section XI, Subsection IWL requirements or corrected in accordance with the corrective action process.

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

1. Examination and acceptance criteria in accordance with the guidance contained in ACI 349.3R.

A.2.1.30 ASME Section XI, Subsection IWF

The ASME Section XI, Subsection IWF aging management program is an existing program that consists of periodic visual examinations of ASME Class 1, 2, and 3 piping and component supports for identification of signs of degradation such as loss of material, loss of mechanical function and loss of pre-load. The inspections are in accordance with American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section XI, Subsection IWF as approved in 10 CFR 50.55(a). The program activities are relied upon to detect and confirm that aging effects of ASME Class 1, 2, and 3 piping and component supports are adequately managed.

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 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."

A.2.1.32 Masonry Wall Program

The Masonry Wall Program is an existing program implemented as part of the Structures Monitoring Program. The Masonry wall condition monitoring is based on guidance provided in IE Bulletin 80-11, "Masonry Wall Design", and IN 87-67, "Lessons Learned from Regional Inspections of Licensee Actions in Response to IE Bulletin 80-11". The Masonry Wall aging management program addresses loss of material, and cracking due to age-related degradation of concrete for masonry walls during the period of extended operation. The program relies on periodic visual inspections to monitor and maintain the condition of masonry walls within the scope of license renewal.

The Masonry Wall Program will be enhanced to include:

1. Add buildings, and masonry walls that have been determined to be in the scope of License Renewal.
    a. Fire Pump House
    b. Masonry Wall Fire Barriers
    c. Office Buildings (The clean and controlled facilities buildings)
    d. SBO Yard Buildings
    e. Service Building
    f. Turbine Building
2. Add an Examination Checklist for masonry wall inspection requirements.
3. Specify an inspection frequency of not greater than 5 years for masonry walls.

These enhancements will be implemented prior to the period of extended operation.

A.2.1.33 Structures Monitoring Program

The Structures Monitoring Program is an existing program that was developed to implement the requirements of 10 CFR 50.65 and is based on NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," and Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,". The program includes the Masonry Wall Program and the RG 1.127, Inspection of Water-Control Structures Associated With Nuclear Power Plants aging management program.

The program relies on periodic visual inspections to monitor the condition of structures and structural components, structural bolting, component supports, masonry block walls, and water control structures. The inspections are conducted on a frequency not greater than 5 years.

The Structures Monitoring Program will be enhanced to include:

1. The scope of the program will be enhanced to include the following structures and components:
    a. Fire Pump House
    b. Office Buildings (The clean and controlled facilities buildings)
    c. SBO Yard Buildings
    d. Service Building
    e. Switchyard
    f. Turbine Building
    g. Transmission towers
    h. Yard Structures (Foundations for fire water and demineralized water tanks, the plant vent radiation monitoring enclosures, the turbine crane runway extensions, and manholes)
    i. Building penetrations and pipe encapsulations that perform flood barrier, pressure boundary, shelter and protection intended functions
    j. Pipe whip restraints and jet impingement/spray shields
    k. Trench covers and sump liners
    l. Masonry walls, including Fire Barriers
    m. Miscellaneous steel (catwalks, vents, louvers, platforms, etc.)
    n. Vortex Suppressor, Ice Barrier, Marine Dock Bumper (Service Water Intake Structure)
    o. Panels, Racks, Cabinets, and Other Enclosures
    p. Metal-enclosed Bus
    q. Components supports including, electrical cable trays, electrical conduit, tubing, HVAC ducts, instrument racks, battery racks, and

supports for piping and components that are not within the scope of ASME Section XI, Subsection IWF

r. Duct banks that contain safety-related cables, and cables credited for SBO or ATWS

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 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.

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.

These enhancements will be implemented prior to entering the period of extended operation.

The following table is provided to tabulate the acceptance criteria from the Structures Monitoring Program Enhancement 5 c. associated with testing the water drained from the Salem Unit 1 SFP telltales and seismic gap drain.

Acceptance Criteria- Salem Unit 1 SFP Telltales and Seismic Gap Drain

Chemical Analysis	Acceptance Criteria		Frequency for monitoring
	SFP Telltales (West Wall)	Seismic Gap Drain (East Wall)	
pH	> 6.0 and < 9.0	> 6.5 and < 10.0	Samples taken monthly
Chloride	≤ 500 ppm	≤ 500 ppm	Samples taken every 6 months
Sulfate	≤ 1,500 ppm	≤ 1,500 ppm	Samples taken every 6 months
Boron	For Information Only	For Information Only	Samples taken monthly
Iron	For Information Only	For Information Only	Samples taken every 6 months

Chemistry results that do not meet one of the criteria will be entered into the corrective action program for an investigation and evaluation.

A.2.1.34 RG 1.127, Inspection of Water-Control Structures Associated With Nuclear Power Plants

The RG 1.127, Inspection of Water-Control Structures Associated With Nuclear Power Plants is implemented through the Structures Monitoring Program. The RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants program is an existing program that will be enhanced to require inspection of water control structures and components that are in scope for license renewal. These structures include the Service Water Intake structure and Shoreline Protection and Dike structures (including outer walls of the Circulating Water Intake Structure). The RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants aging management program addresses age-related deterioration, degradation due to extreme environmental conditions, and the effects of natural phenomena that may affect the safety function of the water control structures. The program manages loss of material, cracking, and change in material properties for concrete components, loss of material and loss of preload for steel and metal components, loss of material and change in material properties for wooden components, hardening and loss of strength for elastomers, and loss of material and loss of form for earthen water control structures. Elements of the program are designed to detect degradations and take corrective actions to prevent a loss of an intended function.

The RG 1.127, Inspection of Water-Control Structures Associated With Nuclear Power Plants Program 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).

These enhancements will be implemented prior to the period of extended operation.

A.2.1.35 Protective Coating Monitoring and Maintenance Program

The Protective Coating Monitoring and Maintenance Program is an existing program that provides for aging management of Service Level I coatings inside the containment structure. Service Level I coatings are used in areas where coating failure could adversely affect the operation of post-accident fluid systems and thereby impair safe shutdown. The Protective Coating Monitoring and Maintenance Program provides for inspections, assessments, and repairs for any condition that adversely affects the ability of Service Level I coatings to function as intended.

A.2.1.36 Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements aging management program is a new program that will be used to manage aging of non-EQ cables and connections during the period of extended operation. A representative sample of accessible cables and connections located in adverse localized environments will be visually inspected at least once every 10 years for indications of accelerated insulation aging such as embrittlement, discoloration, cracking, swelling, or surface contamination. An adverse localized environment is a condition in a limited plant area that is significantly more severe than the specified service environment for the cable or connection.

This new aging management program, including performance of initial inspections, will be implemented prior to the period of extended operation.

A.2.1.37 Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits aging management program 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). This program applies to sensitive instrumentation cable and connection circuits with low-level signals that are in scope for license renewal and are located in areas where the cables and connections could be exposed to adverse localized environments caused by heat, radiation, or moisture. These adverse localized environments can result in reduced insulation resistance causing increases in leakage currents.

Calibration results and findings of surveillance programs for the in-scope portions of the Radiation Monitoring System and the Reactor Protection System will be assessed for cable aging degradation prior to the period of extended operation and at least once every 10 years afterwards.

This new program will be implemented prior to the period of extended operation.

A.2.1.38 Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

The Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements aging management program is a new program that will be used to manage the aging effects and mechanisms of non-EQ, in scope inaccessible medium voltage cables (4,160 volts and 13,800 volts). These cables may at times be exposed to significant moisture simultaneously with significant voltage. Significant moisture is defined as periodic exposures that last more than a few days (e.g., cable in standing water). Periodic exposures that last less than a few days (e.g., normal rain and drain) are not significant. Significant voltage exposure is defined as being subject to system voltage for more than twenty-five percent of the time. Note that no inaccessible medium voltage cable exposed to significant moisture was excluded from the program due to the "significant voltage" criterion. The Salem cables in the scope of this aging management program will be tested using a proven test for detecting deterioration of the insulation system due to wetting that is state-of-the-art at the time the test is performed. 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. The first tests will be completed prior to the period of the extended operation.

Prior to the period of extended operation, 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). The objective of the inspections, as a preventive action, is to minimize the exposure of medium voltage cables to significant moisture. The frequency of inspections for accumulated water will be established based on inspection results. This also recognizes that a recurring inspection, set at the optimum frequency, would result in the cables being submerged only as a result of event driven, rain and drain, type occurrences. Station procedures will direct the assessment of the cable condition as a result of rain or other event driven occurrences. As a limit on the amount of time between inspections, the maximum time between inspections will be no more than 1 year.

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.

This new program, including the enhancement, will be implemented prior to the period of extended operation. In addition, initial cable tests will be implemented prior to the period of extended operation and sufficient manhole and cable vault inspections will be performed prior to the period of extended operation so that proper inspection frequencies are established minimize the exposure of medium voltage cables to significant moisture during the period of extended operation.

A.2.1.39 Metal Enclosed Bus

The Metal Enclosed Bus aging management program is a new condition monitoring program that will manage the aging of in-scope metal enclosed busses at Salem.

The internal portions of the in-scope metal enclosed bus enclosures will be visually inspected for cracks, corrosion, foreign debris, excessive dust build-up and evidence of moisture intrusion. The bus insulation will be visually inspected for signs of embrittlement, cracking, melting, swelling, or discoloration, which may indicate overheating or aging degradation. The internal bus supports will be visually inspected for structural integrity and signs of cracks. Bolted connections are not accessible, but will be checked (sampled) for loose connection using thermography from outside the metal enclosed bus.

Metal enclosed busses are to be free from unacceptable visual indications of surface anomalies, which suggest that conductor insulation degradation exists. In addition no unacceptable indication of corrosion, cracks, foreign debris, excessive dust buildup or evidence of moisture intrusion is to exist. An unacceptable indication is defined as a noted condition or situation that, if left unmanaged, could lead to a loss of intended function. Thermography results will be confirmed to be within the acceptance criteria of administrative program procedures.

This new aging management program will be implemented prior to the period of extended operation. In addition, the first inspections will be completed prior to the period of extended operation and every 10 years thereafter.

A.2.1.40 Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

The Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program is a new program that will be used to confirm 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. The technical basis for the sample selected will be documented. The specific type of test performed will be a proven test for detecting loose connections, such as thermography or contact resistance measurement, as appropriate to the application.

This new aging management program will be implemented and the one-time tests will be completed prior to the period of extended operation.

A.2.2 Plant-Specific Aging Management Programs

This section provides summaries of the plant-specific programs credited for managing the effects of aging.

A.2.2.1 High Voltage Insulators

The High Voltage Insulators program is a new program that manages the degradation of insulator quality due to the presence of salt deposits or surface contamination. This aging effect will be identified through visual inspections of the external surfaces of the high voltage insulators. The visual inspections will be performed on a twice per year frequency.

This new aging management program will be implemented prior to the period of extended operation.

A.2.2.2 Periodic Inspection

The Periodic Inspection aging management program is a new condition-monitoring program that manages the aging of piping, piping components, piping elements, ducting components, tanks and heat exchanger components. This program will manage the aging effects of loss of material, cracking, reduction of heat transfer, and hardening and loss of strength. The program includes provisions for visual inspections of stainless steel, aluminum, copper alloy, and elastomer components not managed under other aging management programs. The program also includes provisions for ultrasonic wall thickness measurements to detect loss of material. Identified deficiencies due to age related degradation are evaluated under the Corrective Action Program.

This new program will be implemented prior to the period of extended operation.

A.2.2.3 Aboveground Non-Steel Tanks

The Aboveground Non-Steel Tanks aging management program is a new program that will manage loss of material of outdoor non-steel tanks in the scope of license renewal. Periodic visual inspections will monitor for degradation of the non-steel tanks external surfaces. Periodic visual inspections will also monitor for degradation of the seal at the interface between the tank bottom and the concrete foundation. Tanks within the scope of this program are the Auxiliary Feedwater Storage Tanks, Primary Water Storage Tanks, Refueling Water Storage Tanks and Demineralized Water Storage Tanks.

The Aboveground Non-Steel Tanks program will include a UT wall thickness inspection of the bottom of the tanks. 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.

This new program, including tank bottom UT inspections, will be implemented prior to the period of extended operation.

A.2.2.4 Buried Non-Steel Piping Inspection

The Salem Buried Non-Steel Piping Inspection program is an existing condition monitoring program that manages buried reinforced concrete piping and components in the Service Water System and Circulating Water System that are exposed to an external soil or groundwater environment for cracking, loss of bond, increase in porosity and permeability, and loss of material. The Salem Buried Non-Steel Piping Inspection aging management program also inspects the buried stainless steel penetration bellows between the Containment Structure and the Fuel Handling Building, including the penetration sleeves, for loss of material. These aging effects will be identified through visual inspections of the external surfaces of the buried piping and components.

The Buried Non-Steel Piping Inspection aging management program 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 years prior to entry into the period of extended operation.

3. Guidance for inspection of concrete aging effects.

These enhancements will be implemented prior to the period of extended operation, with the inspections performed in accordance with the schedule described above.

A.2.2.5 Boral Monitoring Program

The Boral Monitoring Program is an existing program that manages the aging effects of the Boral neutron-absorbing material used in the Exxon and Holtec spent fuel storage rack assemblies in the units 1 and 2 spent fuel pools at Salem. The aging affects that need managing for the Boral neutron-absorbing material during the period of extended operation are reduction of neutron-absorbing capacity and loss of material.

The Boral Monitoring Program performs inspections and/testing on Boral test specimens or coupons. The program monitors changes in physical properties of the Boral by performing measurements on representative Boral test coupons. The Boral test coupons simulate as nearly as possible the actual in-service geometry, physical mounting, materials, and flow conditions of the Boral panels in the spent fuel storage rack assemblies. Monitoring of the Boral neutron-absorbing material is accomplished through periodic examination of the Boral test coupons, consisting of visual observations (which may include photography), and may consist of dimensional measurements (length, width, and thickness), weight and density determinations, and neutron attenuation measurements (for B-10 areal density). The results are evaluated against acceptance criteria for determination of any follow-up activities as appropriate (e.g., removal and examination of additional Boral test coupons, wet chemical analyses, radiography, etc.).

The Boral Monitoring Program 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 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.

These enhancements will be implemented prior to the period of extended operation.

A.2.2.6 Nickel Alloy Aging Management

The Salem Nickel Alloy Aging Management program is an existing program that manages cracking for nickel alloy components in the reactor vessel and steam generators. The Nickel Alloy Aging Management program implements mitigative and condition monitoring activities. Mitigative actions include replacement of components whose materials are susceptible to cracking with materials with improved susceptibility to cracking, and Mechanical Stress Improvement Process on the reactor vessel primary nozzle to safe end welds. The condition monitoring portion of the program uses a number of inspection

techniques to detect cracking due, including surface examinations, volumetric examinations and bare metal visual examinations. The Nickel Alloy Aging Management program implements the inspection of components through an augmented Inservice Inspection program. The augmented program administers component evaluations, examination methods, scheduling, and site documentation as required to comply with regulatory, code, and industry commitments related to nickel alloy issues. The Nickel Alloy Aging Management program implements applicable NRC Bulletins, Generic Letters and staff-accepted industry guidelines.

A.3 NUREG-1801 Chapter X Aging Management Programs

A.3.1 Evaluation of Chapter X Aging Management Programs

Aging Management Programs evaluated in Chapter X of NUREG-1801 are associated with Time-Limited Aging Analyses for metal fatigue of the reactor coolant pressure boundary and environmental qualification (EQ) of electrical components. These programs are evaluated in this section.

A.3.1.1 Metal Fatigue of Reactor Coolant Pressure Boundary

The Metal Fatigue of Reactor Pressure Boundary program is an existing program that manages cumulative fatigue usage for the reactor vessel, the pressurizer, the steam generators, Class 1 and non-Class 1 piping, and Class 1 components subject to the reactor coolant, treated borated water, and treated water environments.

The Metal Fatigue of Reactor Pressure Boundary program is a preventive program that monitors and tracks the number of critical thermal and pressure transients to ensure that the cumulative usage factors for selected reactor coolant system (RCS) components remain less than 1.00 through the period of extended operation. The program determines the number of transients that occur and updates the 60-year projections as required on an annual basis. A software program, WESTEMS, computes cumulative usage factors for select locations.

The effect of the reactor coolant environment on fatigue usage, known as environmental fatigue, has been evaluated for the period of extended operation using the formulae contained in NUREG/CR-6583 for carbon and low-alloy steels and NUREG/CR-5704 for austenitic stainless steels. The fatigue usage associated with the effects of the reactor coolant environment will be included into the ongoing monitoring program.

The program requires the generation of a periodic fatigue monitoring report, including a listing of transient events, cycle summary event details, cumulative usage factors, a detailed fatigue analysis report, and a cycle projection report. If the fatigue usage for any location has had an unanticipated increase based on cycle accumulation trends or if the number of cycles is approaching their limit, the corrective action program is used to evaluate the condition and determine the corrective action. Acceptable corrective actions include repair of the component, replacement of the component, and a more rigorous analysis of the component to demonstrate that the design code limit will not be exceeded during the period of extended operation. Corrective actions include

a review of additional affected reactor coolant pressure boundary locations.

There are several enhancements identified for this existing program as follows.

1. The Metal Fatigue of Reactor Coolant Pressure Boundary program will be enhanced to include additional 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. The Metal Fatigue of Reactor Coolant Pressure Boundary program will be enhanced to use a software program to automatically count transients and calculate cumulative usage on select components.
3. The Metal Fatigue of Reactor Coolant Pressure Boundary program will be enhanced to address 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. The Metal Fatigue of Reactor Coolant Pressure Boundary program will be enhanced to require 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.

These enhancements will be implemented prior to the period of extended operation.

A.3.1.2 Environmental Qualification (EQ) of Electric Components

The Environmental Qualification (EQ) of Electric Components is an existing program that manages the aging of electrical equipment within the scope of 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants." The program establishes, demonstrates, and documents the level of qualification, qualified configurations, maintenance, surveillance and replacements necessary to meet 10 CFR 50.49. A qualified life is determined for components within the scope of the program and appropriate actions such as replacement or refurbishment, or reanalysis, are taken prior to or at the end of the qualified life of the components so that the aging limit is not exceeded. The aging effects are adequately managed so that the intended functions of components within the scope of 10 CFR 50.49 are maintained consistent with the current licensing basis during the period of extended operation.

A.4 Time-Limited Aging Analyses

A.4.1 Introduction

As part of the application for a renewed license, 10 CFR 54.21(c) requires that an evaluation of Time-Limited Aging Analyses (TLAAs) for the period of extended operation be provided. The following TLAAs have been identified and evaluated to meet this requirement.

A.4.2 Reactor Vessel Neutron Embrittlement

The reactor vessel embrittlement calculations for Salem that evaluated reduction of fracture toughness of the Salem reactor vessel beltline materials for 40 years are based upon a predicted End of License fluence of 32 Effective Full Power Years (EFPY). These analyses are considered Time-Limited Aging Analyses (TLAAs) as defined in 10 CFR 54.21(c) and they must be evaluated for the increased neutron fluence associated with 60 years of operation (50 EFPY).

A.4.2.1 Neutron Fluence Analyses

The current reactor vessel embrittlement calculations that evaluated reduction of fracture toughness of the Salem reactor vessel beltline materials for 40 years are based on predicted 40-year EOL fluence values (32 EFPY). These analyses also incorporate the effect of 1.4% power uprate. Therefore, they are Time-Limited Aging Analyses as defined by 10 CFR 54.21(c) and must be evaluated for the increased neutron fluence associated with 60 years of operation.

A value of 50 EFPY for both Salem units was selected to provide conservatism in the fluence projections.

The reactor vessel beltline neutron fluence values applicable to a postulated 20-year license renewal period were calculated for each reactor pressure vessel beltline materials on Salem. The analysis methods used to calculate the Salem vessel fluences satisfy the requirements set forth in Regulator Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence".

The analyses are projected for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A.4.2.2 Upper Shelf Energy Analyses

The current Charpy Upper Shelf Energy (USE) calculations were prepared for each reactor vessel beltline material for Salem based upon projected neutron fluence values for 40 years of service (32 EFPY). These are TLAAs requiring evaluation using 60-year fluence values (50 EFPY).

Appendix G of 10 CFR 50 contains screening criteria that establish limits on how far the USE value for a reactor pressure vessel material may be allowed to decrease due to neutron irradiation exposure. The regulations requires the initial USE value to be greater than 75 ft-lbs in the non-irradiated condition and that the value be greater than 50 ft-lbs in the fully irradiated conditions as determined by Charpy V-notch specimen testing throughout the licensed life of the plant. USE values of less than 50 ft-lbs may be acceptable to the NRC if it can be demonstrated that these lower values will provide margins of safety against brittle fracture equivalent to those required by ASME Section XI, Appendix G.

Per Regulatory Guide 1.99, Revision 2, the Charpy USE should be assumed to decrease as a function of fluence according to Figure 2 of the Regulatory Guide when surveillance data is not used. If surveillance data is used, the decrease in USE may be obtained by plotting the reduced plant surveillance data on Figure 2 of the Regulatory Guide and fitting the data with a line drawn parallel to the existing lines as the upper bound of all of the data. Charpy USE for the beltline forgings and welds were determined using surveillance data (Position 2.2 of the Regulatory Guide), and the Charpy USE for the extended beltline materials was determined without the use of surveillance data (Position 1.2 of the Regulatory Guide).

Predictions of the Charpy USE for EOL (50 EFPY) for Salem have been made using the corresponding 1/4T fluence projection, the copper and nickel content of the beltline materials, and the results of the capsule specimens tested to date, where applicable, using Figure 2 in Regulatory Guide 1.99.

The USE values for the beltline and extended beltline materials are projected to remain above the 50 ft-lb requirement through the period of extended operation.

The analyses are projected for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A.4.2.3 Pressurized Thermal Shock Analyses

10 CFR 50.61(b)(1) provides rules for the protection of pressurized water reactors against pressurized thermal shock. Licensees are required to assess the projected values of nil ductility reference temperature whenever a significant change occurs in the projected values of Reference Temperature Pressurized Thermal Shock (RT_{PTS}), or upon request for a change in the expiration date for the facility operating license. For the current 40-year period, the RT_{PTS} was analyzed for 32 EFPY, which is considered a TLAA.

Reactor vessel beltline fluence is one of the factors used in determining the margin of acceptability of the reactor vessel to pressurized thermal shock as a result of neutron embrittlement. The margin is the difference between the maximum nil ductility reference temperature in the limiting beltline material and the screening criteria established in accordance with 10 CFR 50.61(b)(2). The screening criteria for the limiting reactor vessel materials are 270°F for beltline plates, forgings, and axial weld materials, and 300°F for beltline circumferential weld materials.

The limiting RT_{PTS} value for the Salem Unit 1 axially oriented welds and plates is 258°F, which corresponds to the Lower Shell Longitudinal Weld Seam 3-042C. The limiting RT_{PTS} value for the Salem Unit 1 circumferentially oriented welds and plates is 229°F, which corresponds to the Intermediate-to-Lower Shell Circumferential Weld Seam 9-042.

The limiting RT_{PTS} value for the Salem Unit 2 axially oriented welds and plates is 239°F which corresponds to the Lower Shell Longitudinal Weld Seams 3-442 A&C. The limiting RT_{PTS} value for the Salem Unit 2 circumferentially oriented welds is 118°F, which corresponds to the Intermediate Shell-to-Lower Shell Circumferential Weld Seam 9-442.

Therefore, all of the Salem Units 1 and 2 reactor vessel materials that exceed a surface fluence of 1.0E+17 n/cm2 (E > 1.0 MeV) at 50 EFPY are below the RT_{PTS} screening criteria values of 270°F, for axially oriented welds and plates/forgings, and 300°F, for circumferentially oriented welds, at 50 EFPY.

The analyses are projected for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A.4.2.4 Reactor Vessel Pressure–Temperature Limits, Including Low Temperature Overpressure Protection Limits

Appendix G of 10 CFR 50 requires that the reactor pressure vessel be maintained within established pressure-temperature (P-T) limits, including heatup and cooldown operations. These limits specify the maximum allowable pressure as a function of reactor coolant temperature. As the reactor pressure vessel is exposed to increased neutron irradiation, its fracture toughness is reduced. The P-T limits must account for the anticipated reactor vessel fluence.

The current Low temperature Overpressure Protection (LTOP) setpoint for Salem Units 1 and 2 is 375 psig, and is effective through 32 EFPY for both Units. These calculations are associated with the generation of the P-T limit curves that satisfy the criteria of 10 CFR 54.3(a) and are, therefore, TLAAs.

Updated P-T limits were calculated using fluence values valid for 50 EFPY for Salem reactor vessel beltline region, inlet and outlet nozzles, and closure head flange locations for normal heatup, normal cooldown, and in-service leak and hydrostatic test conditions. In addition, minimum bolt up temperatures, minimum temperature of core criticality, and LTOP system limits were determined. These P-T limits are expressed in the form of a curve of allowable pressure versus temperature.

Salem Units 1 and 2 will submit updates to the P-T and LTOP limits to the NRC at the appropriate time to comply with 10 CFR 50 Appendix G.

The P-T and LTOPs limit analyses will be managed through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

A.4.3 Metal Fatigue of Piping and Components

Metal fatigue was evaluated in the design process for Salem pressure boundary components, including the reactor vessel, reactor coolant pumps, steam generators, pressurizer, piping, valves, and components of primary, secondary, auxiliary, steam, and other systems. The current design analyses for these components have been determined to be Time-Limited Aging Analyses (TLAAs) requiring evaluation for the period of extended operation. Fatigue TLAAs for Salem pressure boundary components are characterized by determining the applicable design code and design specifications that specify the fatigue design requirements.

NUREG-1801 provides a listing of components that are likely to have TLAAs in place that require evaluation for License Renewal. Each of these has been reviewed and the applicable TLAAs are evaluated in the following sections, as appropriate.

In addition, for license renewal, fatigue calculations have been prepared to evaluate the effects of the reactor water environment on the sample of high-fatigue locations applicable to Older-Vintage Westinghouse Plants, identified in Section 5.5 of NUREG/CR-6260. Since several of these components are located within systems currently analyzed to ASA/USAS B31.1 rules, new explicit analyses were prepared in accordance with ASME Section III, Class 1 rules for each of these components. For these locations environmental fatigue correction factors were computed and applied to the CUF values developed in the Class 1 fatigue analyses.

A.4.3.1 Nuclear Steam Supply System (NSSS) Pressure Vessel and Component Fatigue Analyses

Nuclear Steam Supply System (NSSS) pressure vessels and components for Salem were designed in accordance with ASME Section III, Class A or Class 1 requirements, and were required to have explicit analyses of cumulative fatigue usage.

ASME Section III, Class A and Class 1 fatigue analyses are based upon explicit numbers and amplitudes of thermal and pressure transients described in the design specifications. The intent of the design basis transient definitions is to bound not just specific operations but a wide range of possible events with varying ranges of severity in temperature, pressure, and flow. The most limiting numbers of transients used in these NSSS component analyses are considered design limits. Those that are significant contributors to fatigue usage are monitored to assure the limits are not exceeded.

Each Salem Unit 1 and 2 component designed in accordance with ASME Section III, Class A and Class 1 rules was analyzed and shown to have a CUF less than the design limit of 1.0. Since each of the Class A and Class 1 fatigue analyses described above are based upon a number of cycles postulated to bound 40 years of service, they have been identified as TLAAs that require evaluation for 60 years.

The evaluation method used to determine the adequacy of the existing design transients for 60 years operations for the Salem license renewal was as follows:

- An investigation of the actual number of cumulative cycles of each transient type was performed, along with predictions of future cycles, and the results were compared to the original design number of cycles for each transient type to demonstrate that the original design cycles were bounding.
- An investigation of the actual severity of the plant transients in comparison with the severity of the equivalent design basis transients was performed to demonstrate the original design transient severities are bounding.
- The administrative and operating procedures were reviewed in order to assess the effectiveness of the design transient cycle counting program and to validate the cyclic assumptions.

The overall conclusion of these investigations is that the existing design transients are sufficiently conservative for encompassing 60 years of plant operations.

The analyses are valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.4.3.2 Pressurizer Safety Valve and Pilot-Operated Relief Valve Fatigue Analyses

This section is deleted.

A.4.3.3 ASA / USAS B31.1 Piping Fatigue Analyses

Piping designed in accordance with USAS B31.1 Piping Code is not required to have an analysis of cumulative fatigue usage, but cyclic loading is considered in a simplified manner in the design process. When the Salem B31.1 components were designed, the overall number of thermal and pressure cycles expected during the 40-year lifetime of these components was determined. The total number of cycles was compared to cycle ranges specified in USAS B31.1 for consideration of allowable stress reduction. If the total number of cycles exceeded 7,000 cycles, a stress range reduction factor had to be applied to the allowable stress range for secondary stresses (expansion and displacement) to account for thermal cycling. This is considered to be an implicit fatigue analysis since it is based upon a total number of cycles projected to occur in 40 years, but does not have an explicit Cumulative Usage Factor (CUF) value associated with it. Since the overall number of cycles could potentially increase during the period of extended operation, which could potentially result in further reduction of the allowable stress, these implicit fatigue analyses are also considered to be TLAAs requiring evaluation for the period of extended operation.

Note that after originally being designed in accordance with USAS B31.1 requirements, portions of some components of these systems were reevaluated to ASME Section III, Class 1 requirements. The remaining components within those systems that were not reanalyzed are included within the evaluation provided within this section.

In order to evaluate these TLAAs for 60 years, the numbers of cycles expected to occur within the 60-year operational period should be compared to the numbers of cycles that were originally considered in the design of these components. If this number does not exceed 7,000 cycles, the minimum number of cycles required that would result in application of an allowable stress reduction factor, then there is no impact from the added years of service and the original analyses remain valid. If the total number of cycles exceeds 7,000 cycles, then additional evaluation is required.

The 60-year transient projections show that even if all of the projected operational transients are added together, the total number of cycles projected for 60 years will not exceed 7,000 cycles. Therefore, there is no impact upon the implicit fatigue analyses used in the component design for any system that is only affected by operational transients.

The analyses are valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.4.3.4 Supplementary ASME Section III, Class 1 Piping and Component Fatigue Analyses

Each of the Salem piping systems, including the Reactor Coolant System main loop piping, were originally designed in accordance with ASA B31.1-1955 design requirements. Piping systems for Unit 2 were designed in accordance with USAS B31.1.0-1967 requirements. Since then, a number of updated fatigue analyses have been prepared for piping systems and components to address transients that have been identified in the industry that were not originally considered. These analyses have been performed in accordance with ASME Section III, Class 1, rules to enable these transients to be thoroughly evaluated.

Each of these analyses resulted in a Cumulative Usage Factor (CUF) value less than 1.0 based upon a number of transients postulated to bound 40 years of plant operations. Therefore, each of these analyses has been identified as a TLAA requiring evaluation for 60 years.

These analyses are separated from those evaluated in the previous sections because the transient definitions have been modified, or additional transients have been postulated for these components, in addition to those previously described. Therefore, the cycle projections for these components must address these revised transients or additional transient types to determine if they also remain bounded for 60 years of service. Each of these analyses is dispositioned separately within this section for clarity.

A.4.3.4.1 NRC Bulletin 88-08, Thermal Stresses in Piping Connected to Reactor Coolant Systems

NRC Bulletin 88-08 was issued June 22, 1988 with supplements in 1988 and 1989 because of observed pipe cracking due to valve leakage in unisolable lines. The Bulletin required that licensees identify potential locations that might be subject to high stresses due to leaking valves, inspect the potential locations, and to assure that susceptible locations will not fail for the remaining life of the unit.

The NRC Safety Evaluation Report (SER) approved Salem's response to NRC Bulletin 88-08, which included the evaluation of the fatigue analyses of the Normal and Alternate Charging Lines and the Auxiliary Spray Lines. The analyses were based on the requirements of ASME Section III, 1986 edition, Subsection NB-3653 and the fatigue curves of I-9.2.1 and I-9.2.2 and concluded that the cumulative usage factor would remain less than 1.0 for the Normal and Alternate Charging Lines. The Auxiliary Spray Line results for the same transients would remain less than 1.0 for twenty-four (24) calendar years since initial plant start-up. A follow-up analysis was performed and concluded that the fatigue usage for the Auxiliary Spray Line was calculated to be less than 1.0 for a life of 40 calendar years. These analyses are considered TLAAs for evaluation through the period of extended operation.

Normal and Alternate Charging Lines

The fatigue evaluation of the charging lines to address potential thermal cycling transients included typical charging line transients from similar Westinghouse plants designed to ASME Section III, plus additional transients assumed to be induced by valve leakage over a 40 year period, based on operation of either charging line for 60% of the 40-year period, or an equivalent 24-year total leakage period. For the analyses to remain valid for the 60-year period, the cycles of the design transients and those due to leakage must be shown to be less than analyzed. For the design cycles, the fatigue monitoring program can be used to show the charging transient cycles in either the normal or alternate charging line to be less than those analyzed.

A re-evaluation has been performed for the charging nozzles to account for the 60-year projected charging transient cycles, and the resulting fatigue usage is less than 1.0. Although the charging nozzles are not subject to thermal stratification and cycling transients, the results of the re-evaluation demonstrate the relative impact of the 60-year projected charging design transients, and the ability of the design to accommodate these cycles.

The Metal Fatigue of Reactor Coolant Pressure Boundary program will manage the effects of aging due to fatigue in accordance with 10 CFR 54.21(c)(1)(iii).

Auxiliary Spray Line

The Pressurizer Auxiliary Spray line was reanalyzed to ASME Section III, Class 1 design rules in order to evaluate postulated thermal events described in Generic Letter 88-08. The potential thermal events would result from cold water leaking past a closed valve seat into the hot Pressurizer Auxiliary Spray

line, leading to thermal cycling along the bottom of the pipe. Subsequent valve maintenance and monitoring has minimized the likelihood of this type of thermal cycling, but the analysis remains in effect, and was identified as a TLAA requiring evaluation for 60 years.

The 1999 analysis for 40 years of operation concluded that the cumulative usage factor was less than 1.0 for the limiting location of the Auxiliary Spray Line. The basis was five (5) Inadvertent Auxiliary Spray transients. The analysis also stated that the fatigue usage for thermal cycling and striping due to valve leakage was negligible. The 60-yr projected Inadvertent Auxiliary Spray to Pressurizer events are 2 and 3, respectively for Salem Units 1 and 2.

Therefore, the number of projected transients is bounded by the basis for the ASME III fatigue analysis.

The analyses are valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.4.3.4.2 NRC Bulletin 88-11, Pressurizer Surge Line Thermal Stratification

NRC Bulletin 88-11, issued in December 1988, requested utilities to establish and implement a program to confirm the integrity of the pressurizer surge line. The program required both visual inspection of the surge line and demonstration that the design requirements of the surge line are satisfied, including the consideration of stratification effects.

The Pressurizer Surge Line piping and nozzles were previously evaluated for the effects of thermal stratification and plant-specific transients. The controlling fatigue location was the surge line weld to the pressurizer surge nozzle. In a later evaluation, a plant-specific WESTEMS™ model was developed for the pressurizer and surge line to evaluate the effects of pressurizer insurge/outsurge transients and surge line stratification on the pressurizer surge nozzle safe end to pipe weld and the surge line hot leg nozzle, which is the analysis of record and needs to be evaluated as a TLAA for 60 years.

The 60-year analyses demonstrated compliance with design requirements considering ASME Code requirements and utilized the design set of NSSS transients. The Pressurizer Surge Line stratification sub-transients were developed based on plant operating procedures, surge line monitoring data from similar units and historical records for each Salem Unit, and projected 60-year cycles of surge line stratification and insurge/outsurge transients where these were greater than previously evaluated design cycles. These evaluations resulted in CUF less than 1.0 at the pressurizer surge nozzle safe end to pipe weld and at the surge line hot leg nozzle.

The analyses are projected for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A.4.3.4.3 Salem Unit 1 Steam Generator Feedwater Nozzle Transition Piece

As part of the Salem Unit 1 Steam Generators replacement, a new feedwater nozzle transition piece forging was designed as an ASME III, Class 1 component. The specific requirements were ASME III, Subsection NB, 1989. Additionally, the requirements for cyclic operation of Article NB-3200 were incorporated into the design of the transition piece. The remainder of the feedwater piping was designed in accordance with ANSI B31.1.0.

All transients considered in the original design of the feedwater nozzles remain the same, except that the hot standby case was replaced with thermal stratification loadings. This portion of the design assumed 800 hours of auxiliary feedwater flow per cycle, over the course of the fifteen (15) remaining cycles (22 years of balance of life of the plant), resulting in 12,000 hours of auxiliary feedwater operation. This analysis is considered a TLAA for evaluation through the period of extended operation.

The cumulative usage for the Unit 1 Steam Generator Feedwater Nozzle Transition Piece, is based on design basis transients and thermal stratification loads. The cumulative usage will be managed by the Metal Fatigue of Reactor Coolant Pressure Boundary aging management program, where the program will monitor cumulative usage and prevent exceeding its design limit of 1.00.

The Metal Fatigue of Reactor Coolant Pressure Boundary program will manage the effects of aging due to fatigue in accordance with 10 CFR 54.21(c)(1)(iii).

A.4.3.4.4 Salem Unit 1 Steam Generator Primary Manway Studs

The Salem Unit 1 Steam Generator primary manway studs were originally planned for replacement every five (5) years. However, Westinghouse conducted a series of tests to qualify their life for forty (40) years. The analysis conducted for Salem Unit 1 compared the studs installed in the Model F Salem to those studs qualified for extended fatigue life at another Model F plant. The qualification tests were performed in accordance with ASME Code, Section III, Appendix II. This analysis is considered a TLAA for evaluation through the period of extended operation.

Transients were inputs to the fatigue analyses of the studs. The 60-year projected cycles applicable to the Salem Unit 1 Model F Steam Generator primary manway studs are bounded by the cycles in the Westinghouse fatigue analysis.

The analyses are valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.4.3.5 Reactor Vessel Internals Fatigue Analyses

The Salem Reactor Vessel Internals were designed and constructed prior to the development of ASME Code requirements for core support structures, but the reactor coolant system functional design requirements were considered in the design. The Reactor Vessel Internals were implicitly designed for low cycle fatigue based upon the reactor coolant system design transient projections for 40 years, which has been identified as a TLAA.

Post-design analyses consist of two Westinghouse calculation notes; (1) a lower core plate evaluation based on the 1.4% uprate and (2) qualification of the Salem domed lower core support plate, also part of the 1.4% uprate project at Salem.

The two post-design Westinghouse calculation notes are evaluations that analyzed the 1.4% uprate in terms of the effect of changes in significant thermal design transients, which were deemed negligible. Therefore, the cumulative fatigue usage attributable to the significant thermal design transients did not change.

The analyses are valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.4.3.6 Spent Fuel Pool Bottom Plates Fatigue Analyses

Salem responded to the NRC request for additional information (RAI) dated 2/26/96 where the NRC requested an analysis to show that the spent fuel pool (SFP) liner and anchors will not experience significant deformations as a result of thermal loadings. In their RAI, the NRC provided acceptance criteria contained in Tables CC-3720-1 and CC-3730-1 of ASME Section III, Div. 2, 1995. Using a conservative fatigue strength reduction factor of five (5), the Salem analysis determined the maximum alternating stresses and compared them to the austenitic stainless steel curves in the ASME code. The resulting allowable cycles for the bottom liner plates were 1,638 cycles.

A separate analysis evaluated the liner bottom plate for the fuel rack pedestal loads under upset conditions, specifically operating basis earthquake (OBE) loadings. The cumulative usage factor was determined to be 0.00063 for one (1) design basis earthquake (DBE) and twenty (20) OBEs.

The events that would cause full temperature thermal cycles in the SFP are refueling outages which can conservatively be correlated to plant heatups and cooldowns. Since the number of cycles projected to occur in 60 years is well below 1,638 cycles analyzed for the SFP bottom liner, this design analysis remains valid for the period of extended operation.

Since the number of DBEs and OBEs projected to occur in 60 years are below the combination of 1 DBE and 20 OBE cycles analyzed for the SFP bottom liner, this design analysis remains valid for the period of extended operation.

The analyses are valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.4.3.7 Environmentally-Assisted Fatigue Analyses

NUREG-1801, Revision 1, Generic Aging Lessons Learned, contains recommendations on specific areas for which existing programs should be augmented for license renewal. The program description for Aging Management Program X.M1, Metal Fatigue of Reactor Coolant Pressure Boundary Program, provides guidance for addressing environmental fatigue for license renewal. It states that an acceptable program addresses 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. Examples of critical components are identified in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components".

This sample of components can be evaluated by applying environmental life correction factors to the existing ASME Code fatigue analyses using formulae contained in NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low Alloy Steels", and in NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels". Demonstrating that these components have an environmentally adjusted cumulative usage factor less than or equal to the design limit of 1.0 is an acceptable option for managing metal fatigue for the reactor coolant pressure boundary.

NUREG/CR-6260 provided environmental fatigue calculations for an Older Vintage Westinghouse plant using the interim fatigue curves from NUREG/CR-5999 for the locations of highest design CUF for the components listed below:

1. Reactor Vessel Shell and Lower Head

2. Reactor Vessel Inlet and Outlet Nozzles

3. Pressurizer Surge Line (including hot leg and pressurizer nozzles)

4. RCS Piping Charging System Nozzles

5. RCS Piping System Safety Injection Nozzles

6. RHR System Class 1 Piping

For the NUREG/CR-6260 locations identified above, the plant-specific components were identified. ASME fatigue usage factors were calculated for each plant-specific component, and the environmental fatigue (Fen) penalties were applied to obtain the updated fatigue results. The reactor vessel was designed to the requirements of the ASME Code, Section III, explicit fatigue usage factors were available from the design evaluations. The plant specific design fatigue results were used to determine the specific locations to be evaluated and apply the applicable Fen penalties to determine updated fatigue usage with environmentally assisted fatigue (EAF).

Salem piping was designed to ANSI B31.1 Code and there is no original explicit fatigue design. To identify Salem specific piping components to be evaluated, design fatigue calculations for similar components were reviewed to determine limiting component locations with respect to the factors influencing fatigue and considering reactor water environmental effects. For the pressurizer surge line, an ASME fatigue evaluation had been previously performed in response to NRC Bulletin 88-11. However, more detailed evaluations were required to accommodate the Fen penalty factors.

Since there was no explicit fatigue design for Salem, no design or transient specifications to the piping exist. Standard transient descriptions for a 4-loop Westinghouse plant were used as the starting point for the fatigue evaluation. Where Salem specific transient information was available, this was incorporated when applicable.

The evaluations showed that no cumulative usage factors with environmental penalties exceeded 1.0 for 60 years of service for the identified plant-specific locations. Future fatigue evaluations using WESTEMS™ "Design CUF" (NB-3200 module) will include written explanation and justification of any user intervention. Future fatigue design calculations will not use or implement the NB-3600 option (module) of the WESTEMS™ program. The Metal Fatigue of Reactor Coolant Pressure Boundary aging management program (B.3.1.1) will be used to manage the aging effects of environmentally assisted fatigue for the components in Salem LRA Tables 4.3.7-1 and 4.3.7-2.

The Metal Fatigue of Reactor Coolant Pressure Boundary program will manage the effects of aging due to environmentally assisted fatigue in accordance with 10 CFR 54.21(c)(1)(iii).

A.4.4 Other Plant-Specific Analyses

A.4.4.1 Reactor Vessel Underclad Cracking Analyses

Intergranular separations (underclad cracking) in low alloy steel heat-affected zones under austenitic stainless steel weld claddings were first detected in SA-508, Class 2, reactor vessel forgings in 1970. They have been reported to exist in SA-508, Class 2, reactor vessel forgings manufactured to a coarse grain practice and clad by high-heat-input submerged arc processes. The regulatory position regarding this issue is found in Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components."

A detailed analysis of underclad cracks is provided in a topical report in which Westinghouse presented a fracture mechanics analysis to justify the continue operation of all Westinghouse Units for 32 EFPY with underclad cracks in the reactor pressure vessels. The Westinghouse Owners' Group and the NRC identified this topical report as a TLAA that required evaluation for License Renewal.

By letter dated March 1, 2001, as supplemented by letters dated June 15 and July 31, 2001, the Westinghouse Owners' Group submitted another analysis for NRC review. It evaluated the impact of cracks in SA-508 Class 2 and SA-508 Class 3 forgings beneath austenitic stainless steel weld cladding on reactor pressure vessel integrity. The initial NRC SER issued on October 15, 2001 applied only to 3-loop plants. The Westinghouse Owners' Group provided clarification to the NRC in a letter dated June 19, 2002, to include all Westinghouse plants. The NRC issued a revised SER in September 2002 to include all Westinghouse plants and the evaluation was reissued.

The reissued evaluation was used to demonstrate that fatigue growth of the subject flaw is insignificant over 60 years and the presence of the underclad cracks is of no concern relative to the structural integrity of the vessels, since (1) Salem verified that it was bounded by the report, and (2) Salem provided a FSAR summary description of the programs and activities for managing the effects of aging and evaluation of TLAAs for the period of extended operation.

The analyses are valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.4.4.2 Reactor Coolant Pump Flywheel Fatigue Crack Growth Analyses

A Westinghouse report includes a fatigue crack growth analysis that has been identified as a TLAA. The report was submitted for NRC review and the NRC issued a Safety Evaluation Report in September 1996. The purpose of the report was to provide an engineering basis for elimination of flywheel in-service inspection requirements for all operating Westinghouse plants and certain Babcock and Wilcox plants.

The analysis addresses crack growth of a postulated flaw and compares this growth to a critical flaw size to determine whether or not a failure would occur under maximum overspeed conditions. To estimate the magnitude of fatigue crack growth during plant life, an initial radial crack length of 10 percent of the through the flywheel was assumed. The maximum stress intensity range occurs during reactor coolant pump startup. The number of cycles on the flywheel corresponds to the number of reactor coolant pump starts and stops. The number of cycles (pump starts and stops) for a 60-year plant life was assumed to be 6,000 for this analysis. Crack growth was shown to be negligible from exposure to these 6,000 cycles.

The projected number of start/stop cycles for the Salem Units 1 and 2 RCP flywheels are much less than the analyzed 6,000 cycles.

The analyses are valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.4.4.3 Leak-Before-Break Analyses

Appendix A, Criterion 4, of 10 CFR 50 allows for the use of leak-before-break (LBB) methodology for excluding the dynamic effects of postulated ruptures in reactor coolant system piping. The fundamental premise of the LBB methodology is that the materials used in nuclear power plant piping are sufficiently tough that even a large through-wall crack would remain stable and would not result in a double-ended pipe rupture. Application of the LBB methodology is limited to those high-energy fluid systems not considered to be overly susceptible to failure from such mechanisms as corrosion, water hammer, fatigue, thermal aging or indirectly from such causes as missile damage or the failure of nearby components. The analyses involved with LBB are considered TLAAs.

The 60-year LBB analysis demonstrates that the previous LBB conclusions still remain valid, and the dynamic effects of the pipe rupture resulting from postulated breaks in the reactor coolant primary loop piping need not be considered in the Salem structural design basis for the period of extended operation. This 60-year analysis used input from steam generator snubber elimination program, steam generator replacement design change packages, 1.4% power uprate evaluation, the Tavg operating window, and the Mechanical Stress Improvement Process (MSIP) application at the reactor vessel primary nozzle locations.

The analyses are valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.4.4.4 Applicability of ASME Code Case N-481 to the Salem Units 1 and 2 Reactor Coolant Pump Casings

Periodic volumetric inspections of the welds of the primary loop pump casings of commercial nuclear power plants are require by Section XI of the ASME Boiler and Pressure Vessel Code. These inspections require a large amount of time and resources to complete. They also result in large radiation exposure (man-rem). Since the pump casings are inspected prior to being placed in service, and no significant mechanisms exist for crack initiation and propagation, it has been concluded that the in-service volumetric inspection can be replaced with an acceptable alternate inspection. In recognition of this, ASME Code Case N-481, Alternative Examination Requirements for Cast Austenitic Pump Casings, provides an alternative to the volumetric inspection requirement. The code case allows the replacement of volumetric examinations of primary loop pump casings with fracture mechanics-based integrity evaluation (Item (d) of the code case) supplemented by specific visual examinations. Westinghouse demonstrated compliance with ASME Code Case N-481 on a generic basis that is documented in WCAP-13045. In this evaluation, stress analyses were performed to support fracture mechanics analyses for postulated flaws. Salem applied WCAP-13045 to the Salem reactor coolant pump casings for their 40-year plant life.

The TLAA related to Code Case N-481 is thermal aging of cast austenitic stainless steel and its consequence on fatigue crack growth. The 60-year analysis provided a comparison of the Salem pump casing nozzle loadings with the screening loads reported in WCAP-13045. The screening loads in WCAP-13045 bounded the Salem loads anticipated for 60 years of operation. The stability of the flaws postulated in the primary loop pump casings has been established by evaluating the necessary materials properties against the saturated (fully aged) fracture toughness values. The results of the 60-year analysis show that Code Case N-481 is satisfied for the license renewal period when supplemented with the visual inspections specified in the code case (Items a, b, and c).

The analyses are valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.4.4.5 Salem Unit 1 Volume Control Tank Flaw Growth Analysis

Flaws were identified in the shell to lower head weld of the Salem Unit 1 Volume Control Tank (VCT) during 1RF13 (1999). The flaws found during the inspection were subsurface and not in contact with the environment, therefore, only fatigue would be the contributing mechanism to flaw growth. The analyses concluded that an initial flaw would grow an insignificant amount of only $1.1 \times 10^{(-5)}$ inches, based on 1,000 pressurization cycles.

The VCT is an operating surge volume compensating in part for reactor coolant releases from the reactor coolant system as a result of level changes. The major pressurization cycles (transients) experienced by the VCT would be Inadvertent Safety Injection events and Operating Basis Earthquake cycles, and to a lesser extent, Plant Heatups and Cooldowns.

Therefore, since the number of cycles projected to occur in 60 years for Inadvertent Safety Injection events and Operating Basis Earthquake cycles, Plant Heatups, and Plant Cooldowns is well below 1,000 pressurization cycles analyzed for the Unit 1 VCT flaw, this design analysis remains valid for the period of extended operation.

The analyses are valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.4.5 Fuel Transfer Tube Bellows Design Cycles

The fuel transfer tube connects the fuel transfer canal (inside the Containment Structure) to the transfer pool (inside the Fuel Handling Building). The fuel transfer tube passes through the containment wall and through the exterior wall of the Fuel Handling Building.

The fuel handling building fuel transfer tube is comprised of a 24-inch diameter penetration sleeve penetrating through the containment and fuel handling building walls and three (3) sets of expansion joints (bellows). The penetration sleeve and the three bellows perform a water retaining intended function, and are within the scope of license renewal.

Each of these three bellows was designed for a minimum of 50 cycles of seismic movement, therefore, this design analysis is a TLAA requiring evaluation for the period of extended operation.

In order to determine if the design analyses remain valid for 60 years of operation, the number of seismic cycles for 60 years has been conservatively projected. As of January 2009, the Salem transfer tube bellows have been exposed to zero (0) Operating Basis Earthquake cycles. It is anticipated that 2 and 3 Operating Basis Earthquake cycles would occur in 60 years of operation for Salem Units 1 and 2, respectively.

Therefore, since the number of cycles in 60 years is well below the 50 seismic movement cycles analyzed for these bellows, these design analyses remain valid for the period of extended operation.

The analyses are valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.4.6 Crane Load Cycle Limits

A.4.6.1 Polar Gantry Crane

The purchasing specification for the 230/35-ton Polar Crane in the containment structure at Salem required the crane conform to the design requirements of EOCI-61, "Specifications for Electric Overhead Traveling Cranes – 1961. Issuance of the Crane Manufacturers Association of America (CMAA) Specification 70 was meant to supersede EOCI-61. An engineering study reviewed the design of this crane and determined that it complied with CMAA 70, Rev. 75. As such, the design of this crane corresponds to the cyclic loading requirements of CMAA 70, Class A. This evaluation of cycles over the 40-year life is the basis of a safety determination and is therefore a TLAA Analysis.

The Polar Crane was designed for a minimum of 20,000 load cycles, corresponding to the criteria of CMAA Specification 70 for service Class A. The number of anticipated lifts for the Polar Crane is estimated to be 1,720 through the period of extended operation, which is less than the minimum allowable design value of 20,000 cycles.

The analyses are valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.4.6.2 Fuel Handling Crane

The purchasing specification for the 5-ton Fuel Handling Crane in the Fuel Handling Building at Salem required the crane conform to the design requirements of EOCI-61, "Specifications for Electric Overhead Traveling Cranes – 1961. Issuance of the Crane Manufacturers Association of America (CMAA) Specification 70 was meant to supersede EOCI-61. An engineering study reviewed the design of this crane and determined that it complied with CMAA 70, Rev. 75. As such, the design of this crane corresponds to the cyclic loading requirements of CMAA 70, Class A. This evaluation of cycles over the 40-year life is the basis of a safety determination and is therefore a TLAA Analysis.

The Fuel Handling Crane was designed for a minimum of 20,000 load cycles, corresponding to the criteria of CMAA Specification 70 for service Class A. The number of anticipated lifts for the Fuel Handling Crane is estimated at 12,000, which is less than the minimum allowable design value of 20,000 cycles.

The analyses are valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.4.6.3 Cask Handling Crane

The existing Salem Cask Handling Cranes were replaced (Unit 1 in Fall 2008; Unit 2 in Fall 2009) by single failure proof Cask Handling Cranes rated for 115 tons (main hoist) and 10 tons (auxiliary hoist). Each of these two cranes was designed to ASME NOG-1-2004, NUREG-0554, and NUREG-0612 criteria in order to be certified as an NRC-approved single failure proof design. The cranes were also designed to CMAA 70-04 standards for Class A service. This evaluation of cycles over the 40-year life is the basis of a safety determination and is therefore a TLAA Analysis.

The Cask Handling Crane was designed for a minimum of 20,000 load cycles, corresponding to the criteria of CMAA Specification 70 for service Class A. The number of anticipated lifts for the Cask Handling Crane is estimated at 1,560, which is less than the minimum allowable design value of 20,000 cycles.

The analyses are valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.4.7 Environmental Qualification of Electrical Equipment

Thermal, radiation, and cyclical aging analyses of plant electrical and I&C components, developed to meet 10 CFR 50.49 requirements, have been identified as time-limited aging analyses (TLAAs) for Salem. The NRC has established nuclear station environmental qualification (EQ) requirements in 10 CFR 50.49 and 10 CFR 50, Appendix A, Criterion 4. 10 CFR 50.49 specifically requires that an EQ program be established to demonstrate that certain electrical components located in harsh plant environments are qualified to perform their safety function in those harsh environments after the effects of in-service aging. Harsh environments are defined as those areas of the plant that could be subject to the harsh environmental effects of a loss-of-coolant accident (LOCA), high energy line break (HELB), or post-LOCA radiation. 10 CFR 50.49 requires that the effects of significant aging mechanisms be addressed as part of environmental qualification.

Under 10 CFR 54.21(c)(1)(iii), the Salem Environmental Qualification (EQ) of Electric Components program, which implements the requirements of 10 CFR 50.49 (as further defined and clarified by NUREG-0588, and RG 1.89, Rev. 1), is viewed as an aging management program for License Renewal.

Additionally, reanalysis of an aging evaluation to extend the qualifications of components is performed on a routine basis as part of the Salem Environmental Qualification (EQ) of Electric Components program. Important attributes for the reanalysis of an aging evaluation include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, and corrective actions (if acceptance criteria are not met). TLAA demonstration option (iii), which states that the effects of aging will be adequately managed for the period of extended operation, is chosen and the Salem Environmental Qualification (EQ) of Electric Components program will manage the aging effects of the components associated with the environmental qualification TLAA.

NUREG-1800 states that the staff evaluated the industry's EQ program (10 CFR 50.49) and determined that it is an acceptable aging management program to address environmental qualification according to 10 CFR 54.21(c)(1)(iii).

The Environmental Qualification (EQ) of Electric Components program will manage the effects of aging in accordance with 10 CFR 54.21(c)(1)(iii).

A.5 Salem License Renewal Commitment List

Note: This commitment list represents the commitments in effect at the time of the first UFSAR update after issuance of the renewed plant operating license. As such this list is a "point in time" reference document and no changes should be made to this document. The commitments are controlled per the commitment control process as defined in the "Commitment Management Procedure".

* License Renewal Application (LRA)

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA *Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
1	ASME Section XI Inservice Inspection, Sub Sections IWB, IWC, and IWD	Existing program is credited.	A.2.1.1	Ongoing	LRA Section B.2.1.1
2	Water Chemistry	Existing program is credited.	A.2.1.2	Ongoing	LRA Section B.2.1.2
3	Reactor Head Closure Studs	Existing program is credited.	A.2.1.3	Ongoing	LRA Section B.2.1.3
4	Boric Acid Corrosion	Existing program is credited.	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	Existing program is credited.	A.2.1.5	Ongoing	LRA Section B.2.1.5
6	Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)	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	A.2.1.6	Program to be implemented prior to the period of extended operation.	LRA Section B.2.1.6

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA *Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		ferrite. For "potentially susceptible" components, thermal aging embrittlement will be managed through either an enhanced volumetric inspection or a component-specific flaw tolerance evaluation.			
7	PWR Vessel Internals	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
8	Flow-Accelerated Corrosion	Existing program is credited.	A.2.1.8	Ongoing	LRA Section B.2.1.8
9	Bolting Integrity	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	Existing program is credited.	A.2.1.10	Ongoing	LRA Section B.2.1.10
11	Open-Cycle Cooling Water System	Existing program is credited.	A.2.1.11	Ongoing	LRA Section B.2.1.11

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA *Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
12	Closed-Cycle Cooling Water System	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 performance monitoring of selected Chilled Water System components.	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

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA *Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		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.			

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA *Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
13	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems	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. 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.	A.2.1.13	Program to be enhanced prior to the period of extended operation.	LRA Section B.2.1.13
14	Compressed Air Monitoring	Existing program is credited.	A.2.1.14	Ongoing	LRA Section B.2.1.14
15	Fire Protection	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.	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

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA *Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		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.			
16	Fire Water System	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

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (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	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	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

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA *Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		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.			
18	Fuel Oil Chemistry	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. 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.	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

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA *Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		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	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	A.2.1.19	Program to be enhanced prior to the period of extended operation.	LRA Section B.2.1.19

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA *Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		withdrawal of the capsules, in accordance with 10 CFR Part 50, 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.			

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA *Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		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	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, 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.	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 Salem Letter LR-N11-0148 DRAI 3.2.1.48

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA *Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		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.			
21	Selective Leaching of Materials	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 Salem Letter LR-N11-0005 RAI B.2.1.21-01
22	Buried Piping Inspection	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	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 Salem Letter LR-N10-0372

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA *Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		performed prior to entering the period of extended operation to determine soil corrosivity in the 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.			RAI B.2.1.22-02 Salem Letter LR-N10-0444 RAI B.2.1.22-03

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA *Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		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 one (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 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			

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA *Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		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.			
23	One-Time Inspection of ASME Code Class 1 Small Bore-Piping	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	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

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA *Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		• Two Class 1 small-bore socket welds (one per unit) for cracking caused by thermal fatigue (thermal and mechanical loading)			
24	External Surfaces Monitoring	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	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 Components	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting 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.	A.2.1.26	Program to be implemented prior to the period of extended operation.	LRA Section B.2.1.26
27	Lubricating Oil Analysis	Existing program is credited.	A.2.1.27	Ongoing	LRA Section B.2.1.27
28	ASME Section XI, Sub Section IWE	ASME Section XI, Sub Section 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	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.2.1.28-2 Salem Letter LR-N10-0244 RAI 3.5.2.2.1.7-01

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA *Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		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 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			Salem Letter LR-N10-0321 RAI B.2.1.28-04 RAI B.2.1.33-06 Salem Letter LR-N10-0382

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA *Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		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, Section IWE program requirements, to the extent practical			

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA *Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		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. 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 3/4" 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			

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA *Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		condition. • Remove the concrete floor and expose the 1/4" containment liner plate (floor) for a minimum of two of the vertical leak chase channels with holes. Perform examination of exposed 1/4" 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 1/2" 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 1/2" 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 outage, the following specific corrective actions will be performed on Unit 1			

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA *Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		prior to entry into the period of extended operation: • Perform augmented examinations of the 3/4" containment liner (knuckle plate) at 78' elevation in accordance with IWE-2420. • Perform augmented examinations of the areas of the 1/2" containment liner plate behind insulation panels, where loss of material was previously identified, in accordance with IWE-2420. • Remove 1/2" 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, Sub Section IWE program scope will be revised to include the following welds that are currently exempted from Sub Section IWE and governed under ASME Section XI, Sub Section 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.			

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA *Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		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.			
29	ASME Section XI, Sub Section IWL	ASME Section XI, Section 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
30	ASME Section XI, Section IWF	Existing program is credited.	A.2.1.30	Ongoing	LRA Section B.2.1.30
31	10 CFR Part 50, Appendix J	Existing program is credited.	A.2.1.31	Ongoing	LRA Section B.2.1.31
32	Masonry Wall Program	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.	A.2.1.32	Program to be enhanced prior to the period of extended operation.	LRA Section B.2.1.32

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA *Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		2. Add an Examination Checklist for masonry wall inspection requirements. 3. Specify an inspection frequency of not greater than 5 years for masonry walls.			
33	Structures Monitoring Program	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 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	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 Salem Letter LR-N10-0165 RAI B.2.1.33-2 Salem Letter LR-N10-0321 RAI B.2.1.33-05 Salem Letter LR-N10-0414 RAI B.2.1.33-07 Salem letter LR-N11-0041 RAI B.2.1.33-07 update

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA *Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		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			

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		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, Sub Section 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. 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			

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		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.			

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA *Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
34	RG 1.127, Inspection of Water-Control Structures	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	Existing program is credited.	A.2.1.35	Ongoing	LRA Section B.2.1.35
36	Electrical Cables and Connections Not Subject to	Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification	A.2.1.36	Program and initial inspections to be	LRA Section B.2.1.36

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA *Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
	10 CFR 50.49 Environmental Qualification Requirements	Requirements is a new program and will be used to manage aging of non-EQ cables and connections during the period of extended operation.		implemented prior to the period of extended operation.	
37	Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits	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	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	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 Salem Letter LR-N10-0348 LRA Supplement

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA *Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		minimize the exposure of medium voltage cables to significant moisture. The maximum time between inspections will be no longer than one year. 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	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	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	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

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA *Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
42	Periodic Inspection	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	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. 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.	A.2.2.3	Program to be implemented 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.2.2.3
44	Buried Non-Steel Piping Inspection	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	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

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA *Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		extended operation. 3. Guidance for inspection of concrete aging effects.			
45	Boral Monitoring Program	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 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.	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
46	Nickel Alloy Aging Management	Existing program is credited.	A.2.2.6	Ongoing	LRA Section B.2.2.6
47	Metal Fatigue of the Reactor Coolant Pressure Boundary	Metal Fatigue of the Reactor Coolant Pressure Boundary is an existing program that will be enhanced to include:	A.3.1.1	Program to be enhanced prior to the period of extended operation.	LRA Section B.3.1.1

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA *Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		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.			
48	Environmental Qualification of Electric Components (EQ)	Existing program is credited.	A.3.1.2	Ongoing	LRA Section B.3.1.2
49	New P-T Curves	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	Not Applicable	Prior to August 2026	Salem Letter LR-N10-0369 RAI 3.1.1-02

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA *Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		divider plate 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 Salem Letter LR-N10-0438 Revised Response to RAI 3.1.1-03

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA *Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		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	Metal Fatigue of Reactor Coolant Pressure Boundary	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	Not Applicable	Prior to the period of extended operation.	Salem Letter LR-N10-0445 RAI 4.3-08

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA *Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		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.			
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 Salem Letter LR-N11-0057
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 Salem Letter LR-N11-0057