SEABROOK STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

CHAPTER 18
AGING MANAGEMENT PROGRAMS

SECTIONS

APPENDIX

18.1 INTRODUCTION

The renewed operating license is required by 10 CFR 54.21(d) to include an Updated Final Safety Analysis Report (UFSAR) Supplement. The following sections comprise the UFSAR supplement, Chapter 18:

Section 18.1.1	contains a listing of the aging management programs that correspond to NUREG-1801 Chapter XI programs.
Section 18.1.2	contains a listing of the plant-specific aging management programs.
Section 18.1.3	contains a listing of aging management programs that correspond to NUREG-1801 Chapter X programs associated with Time- Limited Aging Analyses (TLAA).
Section 18.1.4	contains a listing of the Time-Limited Aging Analyses (TLAA).
Section 18.1.5	contains a discussion of the Quality Assurance Program and Administrative Controls.
Section 18.1.6	contains a description of corrective action and operating experience program activities
Section 18.2.1	contains a summarized description of the NUREG-1801 Chapter XI programs for managing the effects of aging.
Section 18.2.2	contains a summarized description of the plant-specific programs for managing the effects of aging.
Section 18.2.3	contains a summarized description of the NUREG-1801 Chapter X programs that support the TLAAs.
Section 18.2.4	contains a summarized description of the TLAAs applicable to the period of extended operation.
Appendix 18A	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 system, structures, and components (SSC) 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.

Changes to information within the UFSAR, Chapter 18, will be made in accordance with 10 CFR 50.59.

The License Renewal UFSAR Supplement describes certain programs to be implemented and activities to be completed before the period of extended operation, as follows:

a. Implement those new programs and enhancements to existing programs no later than 6 months prior to the period of extended operation [PEO].

b. Complete those activities by the 6-month date before the PEO or the end of the last refueling outage prior to the PEO, whichever occurs later.

Seabrook Station shall notify the NRC in writing within 30 days after having accomplished item (a) above and include the status of those activities that have been or remain to be completed in item (b) above.

18.1.1 NUREG-1801 Chapter XI Aging Management Programs

The following list of aging management programs correspond to NUREG-1801 Chapter XI programs.

1. ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (18.2.1.1)
2. Water Chemistry (18.2.1.2)
3. Reactor Head Closure Studs (18.2.1.3)
4. Boric Acid Corrosion (18.2.1.4)
5. Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors (18.2.1.5)
6. (Not Used)
7. PWR Vessel Internals (18.2.1.7)
8. Flow-Accelerated Corrosion (18.2.1.8)
9. Bolting Integrity (18.2.1.9)
10. Steam Generator Tube Integrity (18.2.1.10)
11. Open-Cycle Cooling Water System (18.2.1.11)
12. Closed-Cycle Cooling Water System (18.2.1.12)
13. Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems (18.2.1.13)
14. Compressed Air Monitoring (18.2.1.14)
15. Fire Protection (18.2.1.15)
16. Fire Water System (18.2.1.16)
17. Aboveground Steel Tanks (18.2.1.17)
18. Fuel Oil Chemistry (18.2.1.18)
19. Reactor Vessel Surveillance (18.2.1.19)
20. One-Time Inspection (18.2.1.20)
21. Selective Leaching of Materials (18.2.1.21)
22. Buried Piping and Tanks Inspection (18.2.1.22)[1]

[1] Plant-specific aging management program. Program became plant-specific via License Renewal Application supplements.

23. One-Time Inspection of ASME Code Class 1 Small Bore-Piping (18.2.1.23)
24. External Surfaces Monitoring (18.2.1.24)
25. Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (18.2.1.25)
26. Lubricating Oil Analysis (18.2.1.26)
27. ASME Section XI, Subsection IWE (18.2.1.27)
28. ASME Section XI, Subsection IWL (18.2.1.28)
29. ASME Section XI, Subsection IWF (18.2.1.29)
30. 10 CFR 50, Appendix J (18.2.1.30)
31. Structures Monitoring Program (18.2.1.31)
32. Alkali-Silica Reaction (ASR) Aging Management (18.2.1.31A)[1]
33. Building Deformation Aging Management (18.2.1.31B)[1]
34. Electrical Cables and Connections Not Subject to 10 CFR 50.49 EQ Requirements (18.2.1.32)
35. Electrical Cables and Connections Not Subject to 10 CFR 50.49 EQ Requirements Used in Instrumentation Circuits (18.2.1.33)
36. Inaccessible Power Cables Not Subject to 10 CFR 50.49 EQ Requirements (18.2.1.34)
37. Metal Enclosed Bus (18.2.1.35)
38. Fuse Holders (18.2.1.36)
39. Electrical Cable Connections Not Subject to 10 CFR 50.49 EQ Requirements (18.2.1.37)
40. Protective Coating Monitoring and Maintenance (18.2.1.38)

18.1.2 Plant Specific Aging Management Programs

The plant-specific aging management programs are listed below.

1. 345 KV SF_6 Bus (18.2.2.1)
2. Boral Monitoring (18.2.2.2)
3. Nickel Alloy Nozzles and Penetrations (18.2.2.3)

18.1.3 NUREG-1801 Chapter X Aging Management Programs

The following list of aging management programs correspond to NUREG-1801 Chapter X programs associated with Time-Limited Aging Analyses.

1. Metal Fatigue of Reactor Coolant Pressure Boundary (18.2.3.1)

2. Environmental Qualification (EQ) of Electric Components (18.2.3.2)

18.1.4 Time-Limited Aging Analyses Summaries

Summaries of the Time-Limiting Aging Analyses applicable for the period of extended operation are listed below.

1. Neutron Embrittlement of the Reactor Vessel (18.2.4.1)
2. Metal Fatigue of Vessels and Piping (18.2.4.2)
3. Environmental Qualification (EQ) of Electric Components (18.2.4.3)
4. Fatigue of the Containment Liner and Penetrations. (18.2.4.4)
5. Other Plant-Specific TLAAs (18.2.4.5)

18.1.5 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 "*Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants"*. 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 (SSC) that are subject to Aging Management Review (AMR).

18.1.6 Operating Experience

The existing Corrective Action Program and the Operating Experience Program ensure, through the continual review of both plant-specific and industry operating experience, that the license renewal aging management programs are effective to manage the aging effects for which they are credited. The programs are either enhanced or new programs are developed when the review of operating experience indicates that the programs may not be effective. For each aging management program, operating experience is reviewed on a continuing basis. Plant personnel responsible for screening, assigning, evaluating and submitting operating experience are trained to identify and evaluate aging related issues. Evaluation of aging related issues considers potentially affected plant systems, structures, components, materials, environments, aging effects, aging mechanisms and Aging Management Programs.

Aging related program changes, results of inspection activities and evaluation of relevant internal and external operating experience are tracked by the Seabrook Station action tracking/corrective action program.

The operating experience reviews will include evaluation of applicable NUREGS, ISGs, etc., such as future revisions of NUREG-1801, “Generic Aging Lessons Learned (GALL) Report”. Programmatic features such as training of personnel, trending, record retention, self-assessments, etc., will be in accordance with the existing Seabrook Station corrective action and operating experience programs. The Corrective Action Program is part of the Quality Assurance Program, which meets the requirements of 10 CFR Part 50, Appendix B. The Operating Experience Program meets the criteria of NUREG-0737, “Clarification of TMI Action Plan Requirements,” Item I.C.5, “Procedures for Feedback of Operating Experience to Plant Staff,” and interfaces with and relies on active participation in the Institute of Nuclear Power Operations’ operating experience program. Training of plant personnel will be periodic and will account for personnel turnover. Operating experience concerning aging related degradation will be reported to the industry. Any enhancements necessary to fulfill the above criteria will be implemented on an ongoing basis throughout the term of the renewed license.

18.2 AGING MANAGEMENT PROGRAMS

18.2.1 NUREG-1801 Chapter XI Aging Management Programs

18.2.1.1 ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD

American Society of Mechanical Engineer (ASME) Section XI, Subsections IWB, IWC, IWD Inservice Inspection Program facilitates inspections to identify and correct degradation in Class 1, 2, and 3 piping, components, and integral attachments. The program includes periodic visual, surface and/or volumetric examinations of all Class 1, 2 and 3 pressure-retaining components, their supports and integral attachments (including welds, pump casings, valve bodies and pressure-retaining bolting) and leakage tests of pressure retaining components.

The program is implemented in accordance with the requirements of 10 CFR 50.55a, with specified limitations, modifications and NRC-approved alternatives.

18.2.1.2 Water Chemistry

The Water Chemistry Program includes periodic monitoring and control of detrimental contaminants below the levels known to cause cracking, loss of material, or reduction of heat transfer. The primary scope of this program consists of the Reactor Coolant system and related auxiliary systems containing treated water, reactor coolant, treated borated water and steam. The program is based on Electric Power Research Institute (EPRI) PWR primary water chemistry guidelines and Pressurized Water Reactor (PWR) secondary water chemistry guidelines.

18.2.1.3 Reactor Head Closure Studs

The Reactor Head Closure Studs Program conducts inspections of reactor vessel flange stud hole threads, reactor head closure studs, nuts, and washers to manage cracking and loss of material per the requirements of ASME, Boiler and Pressure Vessel Code, Section XI, "*Rules for Inservice Inspection of Nuclear Power Plant Components.*"

18.2.1.4 Boric Acid Corrosion

The Boric Acid Corrosion Program implements the recommendations of Nuclear Regulatory Commission (NRC) Generic Letter 88-05 "*Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants*," and manages aging of structures and components resulting from borated water leakage. The program requires periodic visual inspection of all systems within the scope of license renewal that contain borated water for evidence of leakage, accumulations of dried boric acid, or boric acid damage. The program provides for visual inspections and early discovery of borated water leaks such that mechanical, electrical, and structural components that may be contacted by leaking borated water will not be adversely affected or their intended functions impaired. The program identifies components exhibiting boric acid accumulations or leakage, evaluates the acceptability for continued service of components exhibiting boric acid accumulations or leakage, trends and tracks previously identified leaks or boric acid accumulations and provides corrective actions for the observed leakage sources and any other affected structures and components.

18.2.1.5 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 Program manages the aging effects of cracking due to primary water stress corrosion cracking of the nickel-alloy used in the fabrication of the upper vessel head penetration nozzles. The NRC has approved ASME Code Case N-729-1, *"Examination Requirements for PWR Reactor Vessel Upper Heads with Nozzles Having Pressure-Retaining Partial-Penetration Welds, Section XI, Division"* with conditions in accordance with the requirements of 10 CFR 50.55a (g)(6)(ii)(D). Repair, replacement, and mitigation activities are conducted in accordance with the Seabrook Station ASME Section XI Repair/Replacement program.

18.2.1.6 (NOT USED)

18.2.1.7 PWR Vessel Internals

The PWR Vessel Internals Program implements the guidance provided in EPRI Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227) and EPRI Inspection Standard for PWR Internals (MRP-228).

The program is a condition monitoring program designed to manage the aging effects on the PWR vessel internal components. The recommended activities provided in MRP-227 and additional plant-specific activities not defined in MRP-227 are implemented in accordance with Nuclear Energy Institute (NEI) 03-08, "Guideline for the Management of Materials Issues."

The RVI Inspection Program is based upon the guidance provided in the latest NRC accepted revision of EPRI MRP-227, "EPRI Materials Reliability Program, Pressurized Water Reactor Internals Inspection and Evaluation Guidelines." The RVI Inspection Program is a living program that will be revised as necessary in response to ongoing joint industry efforts aimed at further understanding the aging effects of the RV Internals.

This program is used to manage the effects of age-related degradation mechanisms that are applicable to the PWR vessel internal components. These aging effects include: a) cracking, including stress corrosion cracking (SCC), primary water stress corrosion cracking (PWSCC), irradiation-assisted stress corrosion cracking (IASCC), and cracking due to fatigue/cyclic loading, b) loss of material induced by wear, c) loss of fracture toughness due to either thermal aging or neutron irradiation embrittlement, d) changes in dimensions due to void swelling or distortion, and e) loss of preload due to thermal and irradiation-enhanced stress relaxation or creep.

18.2.1.8 Flow-Accelerated Corrosion

The Flow-Accelerated Corrosion (FAC) Program manages aging effects of loss of material due to wall thinning on the internal surfaces of carbon or low alloy steel piping, elbows, reducers, tees, expanders, and valve bodies containing high energy fluids (both single phase and two phase flow). The program is based on the EPRI guidelines in NSAC-202L, "*Recommendations for an Effective Flow Accelerated Corrosion Program*" and uses the Chexal Horowitz Engineering/ Corrosion Workstation (CHECWORKS) software program as a predictive tool. Included in the FAC program are:

a. an analysis to determine FAC susceptible lines,

b. performance of baseline inspections,

c. follow-up inspections to confirm the predictions,

d. repairing or replacing components, as necessary.

This program also manages wall thinning caused by mechanisms other than FAC in accordance with the guidance provided in LR-ISG-2012-01, "Wall Thinning Due to Erosion Mechanisms".

18.2.1.9 Bolting Integrity

The Bolting Integrity Program manages the aging effects associated with bolting through the performance of periodic inspections for indications of cracking and loss of material and loss of preload. The program also includes repair/replacement controls for ASME Section XI related bolting and generic guidance regarding material selection, thread lubrication and assembly of bolted joints. The program follows the guidelines and recommendations delineated in NUREG-1339,"*Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants*," EPRI NP-5769," *Degradation and Failure of Bolting in Nuclear Power Plants," and* EPRI TR-104213*," Bolted Joint Maintenance & Application Guide*" for comprehensive bolting maintenance.

The Bolting Integrity Program credits other aging management programs for the inspection of bolting. Operator rounds and system walkdowns will also identify joint leakage.

The Bolting Integrity Program credits seven separate aging management programs for the inspection of bolting.

1. ASME Section XI, Inservice Inspection, Subsections IWB, IWC, and IWD provides the requirements for inservice inspection of ASME Class 1, 2, and 3 piping, which includes pressure retaining bolting.

2. ASME Section XI, Subsection IWE Program, includes steel containment shells and their integral attachments.

3. ASME Section XI, Subsection IWF Program, provides the requirements for inservice inspection of ASME Class 1, 2, and 3 component supports.
4. Buried Piping and Tank Program, provides the requirements for the periodic visual inspections of corrosion on buried piping and tanks, including bolting. This program will also manage loss of preload in pressure retaining bolting within the scope of this program by visual inspection for evidence of leakage when the associated piping is inspected by this program.
5. External Surfaces Monitoring Program provides the requirements for the inspection of bolting for steel components such as piping, piping components, ducting and other components within the scope of license renewal.
6. Structures Monitoring Program provides the requirements for the inspection of structural support bolting.
7. Open-Cycle Cooling Water System provides for management of loss of material and loss of preload for service water pump column bolting in raw water environment.

18.2.1.10 Steam Generator Tube Integrity

The Steam Generator Tube Integrity Program manages the aging effects of cracking, loss of material, reduction of heat transfer and wall thinning from flow accelerated corrosion of the Steam Generator components. The program manages the aging of steam generator tubes, tube plugs, tube supports, divider plate assemblies, tube-to-tubesheet welds, heads, primary side tubesheets, and secondary side components that are contained within the steam generator. The program is based on NEI 97-06 Rev. 3, *"Steam Generator Program Guidelines",* and the associated EPRI guidelines, the response and commitment to Generic Letter 97-06, *"Steam Generator Program Guidelines"*, and Seabrook Station Technical Specification 3/4.4.5 *"Steam Generators"* which ensure that the performance criteria for structural integrity, accident-induced leakage, and operational leakage are not exceeded. Seabrook Station has implemented the operational leakage limits found in NUREG-1431, *"Standard Technical Specifications for Westinghouse Pressurized Water Reactors"*. General visual inspections of the internal surfaces of steam generator heads looking for indication of cracking or loss of material (e.g. rust staining) will be performed at least every 72 effective full power months, or every third refueling outage; whichever results in more frequent inspections. This program also utilizes foreign material exclusion and secondary side maintenance activities (e.g. sludge lancing for deposit removal) to minimize component degradation. Technical specification requirements on steam generator tube volumetric examinations, condition monitoring, and operational assessments are performed to ensure tube integrity will be maintained until the next inspection.

Seabrook Station has addressed the potential for cracking of the primary to secondary pressure boundary due to PWSCC of tube-to-tubesheet welds. Amendment 131 to the Facility Operating

License No. NPF-86 for Seabrook Station, Unit No. 1 was issued on September 10, 2012 (ML12178A537). This amendment provides permanent application of steam generator tube alternate repair criteria, H*.

18.2.1.11 Open-Cycle Cooling Water System

The Open-Cycle Cooling Water System Program manages the aging effects of hardening and loss of strength, loss of material, and reduction of heat transfer and loss of coating integrity of internal coatings/linings. This program relies on the implementation of the recommendations of NRC Generic Letter 89-13, "*Service Water System Problems Affecting Safety-Related Equipment.*" The program manages aging effects for components in the circulating water, primary component cooling water, service water, and diesel generator systems.

The program also includes periodic visual inspections of all coatings/linings applied to the internal surfaces of in-scope components where loss of coating or lining integrity could impact the component's and downstream component's current licensing basis intended function. For coated/lined surfaces determined to not meet the acceptance criteria, physical testing is performed where physically possible (i.e., sufficient room to conduct testing) in conjunction with repair or replacement of the coating/lining. The training and qualification of individuals involved in coating/lining inspections of non-cementitious coatings/linings are conducted in accordance with ASTM International Standards endorsed in RG 1.54, Revision 2, as it pertains to the License Renewal in-scope components addressed by LR-ISG-2013-01 (Aging Management of Loss of Coating or Lining Integrity for Internal Coatings/Linings on In-Scope Piping, Piping Components, Heat Exchangers, and Tanks). For cementitious coatings, training and qualifications are based on an appropriate combination of education and experience related to inspecting concrete surfaces.

18.2.1.12 Closed-Cycle Cooling Water System

The Closed-Cycle Cooling Water Program manages aging effects of cracking, loss of material and reduction of heat transfer in closed cycle cooling water systems. Closed-Cycle Cooling Water (CCCW) systems are described as systems not subject to significant sources of contamination, in which water chemistry is controlled and in which heat is not directly rejected to the ultimate heat sink. The program scope includes activities to manage aging in the Primary Component Cooling Water system and Emergency Diesel Generator Jacket Water cooling systems. The program also includes fire pump diesel engine glycol coolant system, the Control Building Air Handling glycol coolant system (safety-related), and the Thermal Barrier Cooling Water system**.** Glycol containing systems within the scope of this program are monitored for the presence of microbiological activity in accordance with the EPRI Closed-Cycle Cooling Water guidelines.

The program includes maintenance of system corrosion inhibitor concentrations to minimize degradation and inspections of opportunity to assess management of component aging. The program is based on the EPRI "*Closed Cycle Cooling Water Chemistry Guidelines*".

18.2.1.13 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 Program manages loss of material of structural components and wear on the rails of lifting systems within the scope of license renewal.

The program employs the use of visual inspections to identify aging effects prior to loss of function. Preventive actions are not associated with these activities.

Only the structural portions of the in-scope cranes and monorails are in the scope of this program.

18.2.1.14 Compressed Air Monitoring

The Compressed Air Monitoring Program manages aging effects of hardening and loss of strength, loss of material, and reduction of heat transfer and assures an oil free, dry air environment in the plant compressed air system, Diesel Generator compressed air subsystem and containment compressed air system components.

This program is in response to NRC GL 88-14 and INPO Significant Operating Experience Report (SOER) 88-01. It also relies on the ASME OM Guide Part 17, and ISA-S7.0.1-1996 as guidance for testing and monitoring air quality and moisture.

A preventative maintenance program encompassing air system component inspection and repair has been in place since the system was initially placed in service. The program includes leak testing (monitoring) of system components.

Seabrook Station committed to maintain instrument air quality in accordance with the Quality Standard for Instrument Air, ISA-S7.3; “Quality Standard for Instrument Air”. Compliance with ISA-S7.3 is verified by continuous monitoring or periodic testing. In-line dew point monitors are used to verify that the dew point of instrument air at the outlet of the instrument air system dryers is at or below a calculated limit. In-line filters are installed which limit air system maximum entrained particle size. These in-line filters meet or exceed the requirements of the quality standard. Periodic replacement of filters is part of the preventative maintenance program for instrument air systems. Air samples are obtained at least annually and tested for to ensure compliance with air quality standards.

18.2.1.15 Fire Protection

The Fire Protection Program manages aging effects to the fire protection and suppression components through detailed inspections. Age-related degradation of the diesel-driven fire pump’s fuel oil supply line is managed through regularly scheduled fire pump performance tests.

The Fire Protection Program includes but is not limited to inspections of fire barrier penetration seals, fire barrier walls, ceilings, floors, fire doors and testing to prove functionality of the diesel driven fire pump fuel oil supply line.

18.2.1.16 Fire Water System

The Fire Water System Program manages the aging effects of loss of material and reduction of heat transfer due to fouling of the Fire Water System components through detailed inspections via the Seabrook Station Surveillance Test Procedures. The program also manages loss of coating integrity of internal coatings/linings.

The Fire Water System Program includes periodic full flow flush tests and system performance testing per the guidance of NFPA 25. The program also includes a visual inspection of the internal surface of the fire protection piping upon each entry to the system for routine or corrective maintenance. These visual inspections will look for loss of material (wall thickness) or changes to the inner diameter of the piping.

The Fire Water System Program is established in accordance with the applicable National Fire Protection Association (NFPA) codes and standards. Full flow testing and visual inspections are conducted to ensure that loss of material due to general, pitting, and crevice corrosion, microbiologically influenced corrosion (MIC), or fouling, and blockage due to fouling is adequately managed.

The chemistry program provides methods and directions for adding various chemicals to various plant systems including the Fire Water Tanks. These chemicals prevent microbiological growth, inhibit scale formation, dispense solids contained in water, improve chlorination efficiency and maintain pH level to prevent corrosion of piping and components.

The program also includes periodic visual inspections of all coatings/linings applied to the internal surfaces of in-scope components where loss of coating or lining integrity could impact the component's and downstream component's current licensing basis intended function. For coated/lined surfaces determined to not meet the acceptance criteria, physical testing is performed where physically possible (i.e., sufficient room to conduct testing) in conjunction with repair or replacement of the coating/lining. The training and qualification of individuals involved in coating/lining inspections of non-cementitious coatings/linings are conducted in accordance with ASTM International Standards endorsed in RG 1.54, Revision 2, as it pertains to the License Renewal in-scope components addressed by LR-ISG-2013-01 (Aging Management of Loss of Coating or Lining Integrity for Internal Coatings/Linings on In-Scope Piping, Piping Components, Heat Exchangers, and Tanks). For cementitious coatings, training and qualifications are based on an appropriate combination of education and experience related to inspecting concrete surfaces.

18.2.1.17 Aboveground Steel Tanks

The Aboveground Steel Tanks Program manages the aging effects of loss of material and cracking on the outside and inside surfaces of aboveground tanks within the scope of License Renewal.

Tanks within the scope of this program include all in-scope outdoor tanks, except fire water storage tanks, constructed on soil or concrete. Indoor large volume storage tanks (greater than 100,000 gallons) designed to near-atmospheric internal pressures, sit on concrete or soil, and exposed internally to water are included within the scope of this program. Tank inside and outside surfaces are inspected by visual, surface, or volumetric examinations as required to detect the applicable aging effect.

The program utilizes the application of protective coatings on the exterior surfaces of the in-scope steel tanks to mitigate corrosion development due to environmental factors. To ensure that the exterior surfaces of the tanks remain protected, the protective coatings, sealant and caulking are visually inspected.

Inspection for degradation of the sealant and caulking will be performed with a visual and tactile examination (manual manipulation) consisting of pressing on the sealant or caulking to detect a reduction in the resiliency and pliability.

Inaccessible locations, such as the tank bottom, will be surveyed by ultrasonic thickness measurements from inside the tank to detect any material degradation. The ultrasonic thickness measurements of fuel oil tanks within the scope of this program will be performed in accordance with the Fuel Oil Chemistry Program.

18.2.1.18 Fuel Oil Chemistry

The Fuel Oil Chemistry Program manages loss of material in the diesel fuel oil systems for the emergency diesel generators, diesel driven fire water pumps and the Auxiliary Boiler fuel oil system through monitoring and maintenance of diesel fuel oil quality.

New fuel oil is sampled and verified to meet the requirements of applicable American Society for Testing and Materials (ASTM) standards D4057 and D2709 prior to offloading to the storage tanks. Stored fuel oil is sampled and verified to meet the requirements of ASTM D2276 or ASTM D4057, and ASTM D2709. The program monitors fuel oil quality and the levels of water in the fuel oil which may cause the loss of material of the tank internal surfaces. The program monitors water and sediment contamination in diesel fuel.

Fuel Oil storage tanks are periodically drained and inspected. This inspection includes ultrasonic thickness measurements of the tank bottom surface to ensure that significant degradation has not occurred. The program also manages loss of coating integrity of internal coatings/linings.

The program also includes periodic visual inspections of all coatings/linings applied to the internal surfaces of in-scope components where loss of coating or lining integrity could impact the component's and downstream component's current licensing basis intended function. For coated/lined surfaces determined to not meet the acceptance criteria, physical testing is performed where physically possible (i.e., sufficient room to conduct testing) in conjunction with repair or replacement of the coating/lining. The training and qualification of individuals involved in coating/lining inspections of non-cementitious coatings/linings are conducted in accordance with

ASTM International Standards endorsed in RG 1.54, Revision 2, as it pertains to the License Renewal in-scope components addressed by LR-ISG-2013-01 (Aging Management of Loss of Coating or Lining Integrity for Internal Coatings/Linings on In-Scope Piping, Piping Components, Heat Exchangers, and Tanks). For cementitious coatings, training and qualifications are based on an appropriate combination of education and experience related to inspecting concrete surfaces.

18.2.1.19 Reactor Vessel Surveillance

The Reactor Vessel Surveillance Program manages the aging effect of loss of fracture toughness due to neutron embrittlement of the low alloy steel Reactor Vessel. The extent of reactor vessel embrittlement for upper-shelf energy and pressure temperature limits for 60 years is projected in accordance with the NRC Regulatory Guide 1.99, "*Radiation Embrittlement of Reactor Vessel Materials.*" The program utilizes the methodology of projecting neutron embrittlement using surveillance data. Monitoring methods are in accordance with 10 CFR 50, Appendix H "*Reactor Vessel Material Surveillance Requirements.*" Testing methods are in accordance with ASTM E 185-82 "*Radiation Embrittlement of Reactor Vessel Materials.*"

18.2.1.20 One-Time Inspection

The One-Time Inspection Program addresses potentially long incubation periods for certain aging effects and provides a means of verifying that an aging effect is either not occurring or is progressing so slowly as to have negligible effect on the intended function of the structure or components. The One-Time Inspection Program provides measures for verifying that an aging management program is not needed, for verifying the effectiveness of an existing program, or for determining that degradation is occurring which will require evaluation and corrective action.

The One-Time Inspection Program includes determination of appropriate inspection sample size, identification of inspection locations, selection of examination technique, specification of acceptance criteria, and evaluation of results to determine the need for additional inspections or other corrective actions. The inspection sample includes locations where the most severe aging effect(s) would be expected to occur. Inspection methods may include visual (or remote visual), surface or volumetric examinations, or other established NDE techniques.

This Program:

- Verifies the effectiveness of the Plant Chemistry Program for managing the effects of aging in portions of piping and components exposed to a treated water environment.
- Verifies the effectiveness of the Fuel Oil Chemistry Program for managing the effects of aging of piping and components in systems that contain fuel oil.
- Verifies the effectiveness of the Lubricating Oil Analysis Program for managing the effects of aging of piping and components in systems that contain lube oil.

- Verification of stainless steel components in the Auxiliary Steam Heating System, to manage the effects of cracking in a steam environment.
- Verification of reinforced fiberglass vinyl ester or bisphenol-A polyester pipe in the Chlorination System, to manage the aging effects of cracking, blistering, and change in color in chlorinated raw water environment.
- For potential loss of material on the internal surfaces of stainless steel Refueling Water Storage Tank, Reactor Makeup Water Storage Tank, and Condensate Storage Tank exposed to treated-water or treated borated water, visual inspections will be performed from inside the tank or volumetric examinations from outside tank using the One-Time Inspection Program. At least 25 percent of the tank's internal surface will be inspected by a method capable of precisely determining wall thickness. The inspection method must be capable of detecting both general and pitting corrosion and must be qualified and demonstrated effective by Seabrook Station.

18.2.1.21 Selective Leaching of Materials

The Selective Leaching of Materials Program manages the aging effects of loss of material in components susceptible to selective leaching that are exposed to condensation, raw water, brackish water, treated water (including closed cycle cooling and steam), or groundwater environment.

The Selective Leaching of Materials Program will include:

1. A one-time inspection of selected components that are susceptible to selective leaching in material/environment combinations where selective leaching has not been previously identified, and
2. Periodic inspections of selected components that are susceptible to selective leaching in material/environment combinations where selective leaching has been previously identified.

Visual inspection and mechanical examination techniques (Brinell hardness testing or other mechanical examination techniques such as destructive testing (when appropriate), scraping, chipping or other types of hardness testing), or additional examination methods that become available to the nuclear industry, will be used to determine if selective leaching is occurring on the surfaces of a selected set of components.

18.2.1.22 **Buried Piping and Tanks Inspection**

The Buried Piping and Tank Inspection Program is a condition monitoring program that manages the aging effects associated with the external surfaces of buried, underground, and inaccessible submerged piping, such as loss of material, cracking and changes in material properties. It addresses piping composed of steel, stainless steel, and polymer. Copper alloy (>15% zinc) components associated with inaccessible submerged Service Water piping are also within the scope of this program. The plant has no buried tanks in scope for license renewal.

Depending on the material, the program includes external coatings, cathodic protection, analyses for soil corrosivity, and quality of backfill as preventive and mitigative actions. The number of inspections is based on the effectiveness of the preventive and mitigative actions. Annual cathodic protection surveys are conducted. Steel components utilizing cathodic protection have an effectiveness acceptance criterion of -850 mV instant off. Inspections are conducted by qualified individuals. The program includes provisions for visual inspections of the protective wraps and coatings on buried steel and stainless steel piping. If damage to the protective wraps or coatings is found and the piping surface is exposed, the pipe is inspected for loss of material due to general, pitting, crevice or microbiologically-influenced corrosion. If corrosion has occurred, the wall thickness will be determined. Steel and stainless steel piping will be inspected for stress corrosion cracking using volumetric non-destructive examination techniques. Polymer piping is inspected for changes in material properties and for indication of cracking and blistering. Where the coatings, backfill or the condition of exposed piping does not meet acceptance criteria such that the depth or extent of degradation of the base metal could have resulted in a loss of pressure boundary function when the loss of material rate is extrapolated to the end of the period of extended operation, an increase in the sample size is conducted. If a reduction in the number of inspections recommended in the Buried Piping and Tank Inspection Aging Management Plan, is claimed based on a lack of soil corrosivity, as determined by soil testing, then soil testing is conducted once in each 10-year period starting 10 years prior to the period of extended operation. The program includes verification of the effectiveness of the cathodic protection system, non-destructive evaluation of the pipe wall thicknesses, pressure testing of the pipe, internal inspections, and monitoring of the fire protection system jockey pump operation.

This program also manages the aging effects (loss of material and loss of preload) of buried, underground, or inaccessible submerged piping system bolting.

18.2.1.23 <u>One-Time Inspection of ASME Code Class 1 Small Bore-Piping</u>

The One-Time Inspection of ASME Code Class 1 Small Bore Piping Program applies to small-bore ASME Code Class 1 piping less than 4 inches nominal pipe size (NPS), including pipe, fittings, and branch connections. While the ASME Boiler and Pressure Vessel Code, Section XI, *"Rules for Inservice Inspection of Nuclear Power Plant Components"*, does not require volumetric examination of Class 1 small-bore piping, the Seabrook Station One-Time Inspection of ASME Code Class 1 Small Bore Piping Program will be used to identify cracking by performing volumetric examinations of selected piping.

The inspection sample determination will include both socket welds and butt welds. If no demonstrated method of non-destructive volumetric examination capable of detecting cracking in socket welded piping is available, Seabrook Station will remove the selected weld(s) for destructive examination.

18.2.1.24 External Surfaces Monitoring

The External Surfaces Monitoring Program manages aging effects through visual inspection of external surfaces for evidence of hardening and loss of strength, reduction of heat transfer, cracking due to stress corrosion cracking, and loss of material (galvanic, general, crevice and pitting corrosion, and wear). This program consists of periodic inspections of aluminum, Cast Austenitic Stainless Steel (CASS), copper alloy, copper alloy >15% zinc, elastomer, galvanized steel, gray cast iron, nickel alloy, stainless steel and steel components such as piping, piping components, ducting, pipe supports and other components to manage aging effects.

The External Surfaces Monitoring Program utilizes periodic plant system inspections and walkdowns to monitor for materials degradation and leakage. This program inspects components such as piping, piping components, ducting and other components, including bolting. Coatings deterioration is monitored as an indication of possible underlying degradation.

The External Surfaces Monitoring Program includes visual inspections on insulation jacketing to ensure that no aging effects are impairing the function of the thermal insulation. The External Surfaces Monitoring Program also includes periodic inspections of in-scope insulated components for possible corrosion under insulation. A sample of outdoor component surfaces that are insulated and a sample of indoor insulated components exposed to condensation (due to the in-scope component being operated below the dew point), will be periodically inspected every 10 years during the period of extended operation. Subsequent inspections will consist of examination of the exterior surface of the insulation for indications of damage to the jacketing or protective outer layer of the insulation if the initial inspection verifies no loss of material or cracking. If the external visual inspections of the insulation reveal damage to the exterior surface of the insulation or there is evidence of water intrusion through the insulation (e.g. water seepage through insulation seams/joints), periodic inspections under the insulation will continue as conducted for the initial inspection.

18.2.1.25 Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components

The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program manages the aging effects of cracking, loss of material, fouling, reduction of heat transfer, blistering, hardening and loss of strength, and loss of coating integrity of internal coatings/linings. This program consists of inspections of the internal surfaces of aluminum, CASS, copper alloy, copper alloy >15% zinc, elastomer, galvanized steel, gray cast iron, nickel alloy, fiberglass, stainless steel, steel piping, piping components, ducting, and other components that are not covered by other aging management programs.

The program inspections are inspections of opportunity, performed during pre-planned periodic system and component surveillances or during maintenance activities when the systems are opened and the surfaces made accessible for visual inspection. This maintenance may occur during power operations or refueling outages when many systems are opened. Inspections of opportunity are supplemented with focused inspections to ensure that a representative sample of 20% or a maximum of 25 components of each identified material, environment, and aging effect combinations are inspected in each 10 year period during the period of extended operation. Where practical, the population to be inspected is selected from components most susceptible to aging because of time in service and severity of operating conditions. Opportunistic inspections continue in each period despite meeting the sampling limit. The visual inspections assure that existing environmental conditions are not causing material degradation that could result in a loss of the component intended function.

The program also includes periodic visual inspections of all coatings/linings applied to the internal surfaces of in-scope components where loss of coating or lining integrity could impact the component's and downstream component's current licensing basis intended function. For coated/lined surfaces determined to not meet the acceptance criteria, physical testing is performed where physically possible (i.e., sufficient room to conduct testing) in conjunction with repair or replacement of the coating/lining. The training and qualification of individuals involved in coating/lining inspections of non-cementitious coatings/linings are conducted in accordance with ASTM International Standards endorsed in RG 1.54, Revision 2, as it pertains to the License Renewal in-scope components addressed by LR-ISG-2013-01 (Aging Management of Loss of Coating or Lining Integrity for Internal Coatings/Linings on In-Scope Piping, Piping Components, Heat Exchangers, and Tanks). For cementitious coatings, training and qualifications are based on an appropriate combination of education and experience related to inspecting concrete surfaces.

18.2.1.26 Lubricating Oil Analysis

The Lubricating Oil Analysis Program obtains and analyzes lubricating oil samples from plant equipment to ensure that the oil quality is maintained within established limits. The program provides an early indication of adverse equipment condition in lubricating oil environments.

The Seabrook Station Lubricating Oil Analysis Program includes sampling and analysis of lubricating oil for components within the scope of license renewal and subject to aging management review, that are exposed to lubricating oil and for which pressure boundary integrity or heat transfer is required for the component to perform its intended function. The lube oil analysis required will include "Flash Point" when there is a potential for contamination of the lubrication oil by fuel.

18.2.1.27 ASME Section XI, Subsection IWE

The ASME Section XI, Subsection IWE Program manages aging effects to the containment liner, electrical penetrations, mechanical penetrations (piping, ventilation, and spares), personnel lock, equipment hatch, recirculation sump, reactor pit, moisture barriers, seals, gaskets, and supports.

The program performs inspections using the same primary Inservice Inspection method as specified in ASME Section XI, Subsection IWE "*Requirements for Class MC and Metallic Liners* *of Class CC Components of Light-Water Cooled Power Plants* "; visual examination (general visual, VT-3, VT-1).

18.2.1.28 ASME Section XI, Subsection IWL

The ASME Section XI, Subsection IWL, Inservice Inspection Program manages aging effects to the steel reinforced concrete for the containment building and complies with the requirement of examination requirements of 10 CFR 50.55a in accordance with ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWL "*Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants*".

The primary inspection methods used are VT-1C visual examination, VT-3C visual examination and alternative examination methods (in accordance with IWA-2240). All accessible containment reinforced concrete components are within the scope of this program.

To manage the aging effects of cracking due to expansion and reaction with aggregates in concrete structures, the existing ASME Section XI, Subsection IWL Program, 18.2.1.28, and the Structures Monitoring Program, 18.2.1.31, have been augmented by the plant specific Alkali-Silica Reaction (ASR) Monitoring Program, 18.2.1.31A.

18.2.1.29 ASME Section XI, Subsection IWF

The ASME Section XI, Subsection IWF "*Requirements for Class 1,2,3, and MC Component Supports of Light-Water Cooled Power Plants*," Inservice Inspection Program (ISI) provides inspections of Class 1, 2, and 3 Component Supports. For supports other than piping supports, the supports of only one component of a group having similar design, function, and service must be examined. Supports of piping and other items exempted from volumetric or surface examination are also exempt.

The program uses VT-3 visual examination for detection of degradation. The performance requirements for VT-3 examination are conducted to determine the general mechanical and structural condition of components and their supports.

18.2.1.30 10 CFR 50, Appendix J

The 10 CFR Part 50, Appendix J Program implements Title 10 Code of Federal Regulations Part 50 Appendix J, "*Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors*" Option B. The test requirements of Appendix J provide for periodic verification by tests of the leak-tight integrity of the primary reactor containment. The purposes of the tests are to assure that 1) leakage through the containment or systems and components penetrating the containment does not exceed the allowable leakage rate specified in the Technical Specifications and Updated Final Safety Analysis Report, and 2) integrity of the containment structure is maintained during its service life.

18.2.1.31 Structures Monitoring Program

The Structures Monitoring Program includes the Masonry Wall Program and the Inspection of Water Control Structures Associated with Nuclear Power Plants Program.

The Structures Monitoring Program is implemented through the plant Maintenance Rule Program, which is based on the guidance provided in NRC Regulatory Guide 1.160 "*Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*" and NUMARC 93-01 "*Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*", and with guidance from ACI 349.3R, "Evaluation of Existing Nuclear Safety-Related Concrete Structures". The Structures Monitoring Program was developed using the guidance of these three documents. The Program is implemented to monitor the condition of structures and structural components within the scope of the Maintenance Rule, such that there is no loss of structure or structural component intended function.

18.2.1.31A Alkali-Silica Reaction (ASR) Aging Management

The plant specific ASR Aging Management Program manages cracking due to expansion and reaction with aggregates of concrete structures within the scope of license renewal. The potential impact of ASR on the structural strength and anchorage capacity of concrete is a consequence of strains resulting from the expansive gel.

The Structures Monitoring Program and Section XI Subsection IWL Program perform visual inspections of the concrete structures at Seabrook Station for indications of the presence of alkali-silica reaction (ASR). ASR involves the formation of an alkali-silica gel which expands when it absorbs water. This expansion is volumetric in nature but is most readily detected by visual observation of cracking on the surface of the concrete. This cracking is the result of expansion that is occurring in the in-plane directions. Expansion is also occurring perpendicular (through the thickness of the wall) to the surface of the wall, but cracking will not be visible in this direction from the accessible surface. Cracking on the surface of the concrete is typically accompanied by the presence of moisture and efflorescence. Concrete affected by expansive ASR is typically characterized by a network or "pattern" of cracks. Micro-cracking due to ASR is generated through forces applied by the expanding aggregate particles and/or swelling of the alkali-silica gel within and around the boundaries of reacting aggregate particles. The ASR gel may exude from the crack forming white secondary deposits at the concrete surface. The gel also often causes a dark discoloration of the cement paste surrounding the crack at the concrete surface. If "pattern" or "map" cracking typical of concrete affected by ASR is identified, an evaluation will be performed to determine further actions.

ASR is primarily detected by non-intrusive visual observation of cracking on the surface of the concrete. The cracking is typically accompanied by the presence of moisture and efflorescence. ASR may also be detected or confirmed by removal of concrete cores and subsequent petrographic analysis.

Monitoring of crack growth is used to assess the in-plane expansion associated with ASR and to specify monitoring intervals. A Combined Cracking Index (CCI) is established at thresholds at which structural evaluation is necessary (see table below). The Cracking Index (CI) is the summation of the crack widths on the horizontal or vertical sides of a 20-inch by 30 -inch grid on the ASR-affected concrete surface. The horizontal and vertical Cracking Indices are averaged to obtain a Combined Cracking Index (CCI) for each area of interest. A CCI of less than the 1.0 mm/m can be deemed acceptable with deficiencies (Tier 2). Deficiencies determined to be acceptable with further review are trended for evidence of further degradation. The change from qualitative monitoring to quantitative monitoring occurs when the Cracking Index (CI) of the pattern cracking equals or is greater than 0.5 mm/m in the vertical and horizontal directions. Concrete crack widths less than 0.05 mm cannot be accurately measured and reliably repeated with standard, visual inspection equipment. A CCI of 1.0 mm/m or greater requires structural evaluation (Tier 3). All locations meeting Tier 3 criteria will be monitored for in-plane expansion (Via CCI), through-thickness expansion (via borehole extensometer), and volumetric expansion (using CCI and extensometer measurements) on a ½ year (6-month) inspection frequency. All locations meeting the Tier 2 structures monitoring criteria will be monitored on a 2.5 year (30 month) frequency. CCI correlates well with strain in the in-plane directions and the ability to visually detect cracking in exposed surfaces making it an effective initial detection parameter. In the event ASR monitoring results indicate a need to amend either the monitoring program acceptance criteria or the frequency of monitoring; Seabrook Station will take such action under the Operating Experience element of the Alkali-Silica Reaction Aging Management Program.

Tier	Structural Monitoring Program Category	Recommendation for Individual Concrete Components	Criteria
3	Unacceptable (requires further evaluation)	• Structural Evaluation • Implement enhanced ASR monitoring, such as through-wall expansion monitoring using Extensometers.	1.0 mm/m (0.1%) or greater strain measurement (CCI or pin-pin)
2	Acceptable with Deficiencies	Quantitative Monitoring and Trending	• 0.5 mm/m (0.05%) or greater strain measurement (CCI or pin-pin) • CI or pin-pin measurement of greater than 0.5 mm/m (0.05%) in the vertical and horizontal directions
		Qualitative Monitoring	Any area with visual presence of ASR (as defined in FHWA-HIF-12-022) accompanied by a CI of less than 0.5 mm/m (0.05%) in the vertical and horizontal directions.
1	Acceptable	Routine inspection as prescribed by the Structural Monitoring Program	Area has no indications of pattern cracking or water ingress - No visual symptoms of ASR

The Alkali-Silica Aging Management Program was initially based on published studies describing screening methods to determine when structural evaluations of ASR affected concrete are appropriate. Large-scale destructive testing of concrete beams with accelerated ASR has confirmed that parameters being monitored are appropriate to manage the effects of ASR and that an acceptance criterion of 1 mm/m provides sufficient margin with regard to the effect of ASR expansion on structural capacity.

For heavily reinforced structures, in-plane expansion is limited. In-plane expansion measurements (i.e., CCI and embedded pin measurements) were observed in the large-scale test programs to plateau at a relatively low level of accumulated strain. While in-plane expansion monitoring (i.e., CCI and embedded pins) remains useful for the detection and monitoring of ASR at the initial stages, an additional monitoring parameter in the out-of-plane direction is required to monitor more advanced ASR progression. ASR expansion in the out-of-plane direction will be monitored by borehole extensometers installed in drilled core bore holes. In the selected locations, cores have and will continue to be removed for modulus testing to establish the level of through-thickness expansion to date. Instruments (extensometers) have and will continue to be placed in the resulting bore holes to monitor expansion in this direction going forward. The measured in-plane expansion and through-thickness expansion are used to determine volumetric expansion. Expansion measurements are used to maintain the limits specified below.

Structural Design Issue	Criteria[1]
Flexure & reinforcement anchorage	See FP#101020 - Section 2.1 for limit on through-thickness expansion
Shear	See FP#101050 – Appendix B for limit on volumetric expansion
Anchor bolts and structural attachments	See FP#101020 - Section 2.1 for limit on in-plane expansion

Seabrook Station has and will continue to perform several actions to confirm that expansion behavior at the plant is consistent with the specimens from the large-scale test programs. These actions, described in the table below, assess similarity of expansion behavior in terms of trends between directions and expansion levels. These actions also include corroborating the correlation of normalized modulus versus through-thickness expansion derived from the large-scale test programs against plant data.

[1] Expansion Limit Criteria is considered proprietary to NextEra Energy Seabrook. FP #101020 MPR-4288, Revision 0, “Seabrook Station: Impact of Alkali-Silica Reaction on the Structural Design Evaluations,” July 2016; FP#101050 MPR-4273, Revision 1, “Seabrook Station – Implications of Large-Scale Test Program Results on Reinforced Concrete Affected by Alkali-Silica Reaction,” March 2018; License Amendment Request 16-03, “Revise Current Licensing Basis to Adopt a Methodology for the Analysis of Seismic Category I; Structures with Concrete Affected by Alkali-Silica Reaction,” August 1, 2016.

Periodic Confirmation of Expansion Behavior		**When**
Lack of mid-plane crack	Review of records for cores removed to date or since last assessment	Periodic assessments: At least 5 years prior to the Period of Extended Operations (PEO) Every 10 years thereafter
Expansion initially similar in all directions but becomes preferential in z-direction	Compare measured in-plane expansion (εxy) to through-thickness expansion (εz) using a plot of εz versus Combined Cracking Index (CCI)	
Expansions within range observed in test programs	Compare measured εxy, εz and εv (volumetric expansion) at the plant to limits from test programs to check margin for future expansion	
Corroborate modulus-expansion correlation with plant data (A secondary objective of these studies is to provide additional data to confirm that expansion behavior at the plant is comparable to the test specimens.)	For 20% of the 3 extensometer locations: • Remove cores for modulus Compare $\Delta\varepsilon z$ determined from the modulus-expansion correlation with $\Delta\varepsilon z$ determined from the extensometer and the original modulus result	At least 5 years prior to PEO (initial study) and 10 years thereafter (follow-up study). A detailed explanation of this approach is provided in MPR-4273, Revision 1, “Seabrook Station - Implications of Large-Scale Test Program Results on Reinforced Concrete Affected by Alkali-Silica Reaction” (Seabrook FP# 101050).

18.2.1.31B Building Deformation Aging Management

The Building Deformation Aging Management Program is a plant specific program implemented under the existing Maintenance Rule Structures Monitoring Program. Building Deformation is an aging mechanism that may occur as a result of other aging effects of concrete. Building Deformation at Seabrook Station is primarily a result of the alkali silica reaction (ASR) but can also result from swelling, creep, and shrinkage. Building deformation can cause components within the structures to move such that their intended functions may be impacted.

The Building Deformation Aging Management Program uses visual inspections associated with the Structures Monitoring Program and cracking measurements associated with the Alkali-Silica Reaction program to identify buildings that are experiencing deformation. The first inspection is a baseline to identify areas that are exhibiting surface cracking. The surface cracking will be characterized and analytically documented. This inspection will also identify any local areas that are exhibiting deformation. The extent of surface cracking will be input into an analytical model. This model will determine the extent of building deformation and the frequency of required visual inspections.

For building deformation, location-specific measurements (e.g. via laser target and gap measurements) will be compared against location-specific criteria to evaluate acceptability of the condition.

Structural evaluations are performed on buildings and components affected by deformation as necessary to ensure that the structural function is maintained. Evaluations of structures will validate structural performance against the design basis, and use results from the large-scale test programs, as appropriate.

Evaluations for structural deformation consider the impact to functionality of affected systems and components (e.g. conduit expansion joints). Seabrook Station will evaluate the specific circumstances against the design basis of the affected system or component. Structural evaluations will be used to determine whether additional corrective actions (e.g., repairs, additional inspections and/or analysis) to the concrete or components are required. Specific criteria for selecting effective corrective actions will be evaluated on a location-specific basis.

18.2.1.32 Electrical Cables and Connections Not Subject to 10 CFR 50.49 EQ Requirements

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program manages the aging of accessible non-EQ cables and connections. Accessible cables and connections located in adverse localized environments shall be visually inspected 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. Accessible cables and connections shall be inspected prior to entering the period of extended operation, and at least once every 10 years thereafter.

18.2.1.33 Electrical Cables and Connections Not Subject to 10 CFR 50.49 EQ Requirements Used in Instrumentation Circuits

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program manages the aging of in-scope cables and connections. The 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.

The program shall perform insulation resistance tests on the in-scope neutron flux monitoring cable and connections in the Nuclear Instrumentation System.

The frequency of the tests on these cables shall be based on engineering evaluation, but the test frequency shall be at least once every ten years. The first test shall be performed prior to entering the period of extended operation.

18.2.1.34 <u>Inaccessible Power Cables Not Subject to 10 CFR 50.49 EQ Requirements</u>

The Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program manages the aging of inaccessible ≥ 400 volt power cables exposed to adverse localized environments caused by significant moisture. Seabrook Station defines significant moisture as periodic exposures to moisture that last more than a few days (e.g., cable in standing water). Seabrook Station considers periodic exposures to moisture that last less than a few days (i.e., normal rain and drain) as not being significant.

The program includes the following two components:

- **Periodic Inspections of Manholes Containing In-Scope Power Cables**

  In-scope manholes shall be periodically inspected for water collection. Water found in the manholes shall be drained.

  The frequency of manhole inspections shall be based on plant specific operating experience with cable wetting or submergence (i.e. the inspection is performed periodically based water accumulation over time and event driven occurrences, such as heavy rain or flooding). However, the maximum time between inspections shall be no more than one year. The first inspections shall be performed prior to entering the period of extended operation.

- **Testing of In-Scope Inaccessible Power Cables**

  The specific type of test performed shall be determined prior to the initial test, and shall be a proven test for detecting deterioration of the insulation system due to wetting, such as power factor, partial discharge, or polarization index, as described in EPRI guidelines for "*Effects of Moisture on the Life of Power Plant Cables*" or other testing that is state-of-the-art at the time the test is performed. Cable testing shall be performed prior to entering the period of extended operation and at least once every 6 years thereafter.

18.2.1.35 Metal Enclosed Bus

The Metal Enclosed Bus Program manages the aging of in-scope metal enclosed buses. The internal portions of the in-scope metal enclosed bus enclosures are inspected for cracks, corrosion, foreign debris, excessive dust buildup and evidence of moisture intrusion. The bus insulation is visually inspected for signs of embrittlement, cracking, melting, swelling, or discoloration, which may indicate overheating or aging degradation. The internal bus supports are visually inspected for structural integrity and signs of cracks. Accessible bolted bus connections are checked for looseness using thermography from outside the metal enclosed bus.

The inspections and tests shall be performed prior to entering the period of extended operation and at least once every 10 years thereafter.

Aging management of the Metal Enclosed Bus enclosures and elastomers is included in the Structures Monitoring Program.

18.2.1.36 Fuse Holders

The Fuse Holders Program manages the aging of in-scope metallic clamps of fuse holders.

The Fuse Holders Program performs tests on in-scope fuse holders (metallic clamps). The type of test is a proven test, such as thermography or contact resistance which detects thermal fatigue in the form of high resistance caused by corrosion or oxidation. The type of test performed is determined prior to the initial test. The first test shall be performed prior to entering the period of extended operation and at least once every 10 years thereafter.

18.2.1.37 Electrical Cable Connections Not Subject to 10 CFR 50.49 EQ Requirements

The Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is a one-time testing program that shall be used to verify the absence of aging effects on the metallic portion of electrical cable connections. The aging effect and mechanism of concern is the loosening of bolted connections due to thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, and oxidation.

The scope of this sampling program considers application (medium and low voltage), circuit loading (high loading), and location (high temperature, high humidity, vibration, etc).

The specific type of test performed is a proven test for detecting loose connections, such as thermography or contact resistance measurement, as appropriate for the application.

The one-time test shall be performed prior to entering the period of extended operation.

18.2.1.38 Protective Coating Monitoring and Maintenance

The Protective Coating Monitoring and Maintenance program manages cracking, blistering, flaking, peeling, and delamination of the Service Level I coatings consistent with the guidelines of

Regulatory Position C4 of the Nuclear Regulatory Commission [NRC] Regulatory Guide [RG] 1.54, Rev. 1) as described in NUREG 1801, Rev. 1.

At the beginning of every refueling outage, the Seabrook Station coating supervisor and the design engineer shall inspect all areas and components from which peeling coatings have the potential of falling into the reactor cavity or CBS [Emergency Core Cooling System (ECCS)] recirculation sumps. These areas and components shall include but not be limited to the following as applicable: polar crane, refueling machine, manipulator crane, CRDM cooling fan shrouds, wall and equipment adjacent to reactor cavity, carbon steel supports and hangers within the reactor cavity.

The determination of acceptability of the coatings will be made by the Design Engineer. The inspection report is to be evaluated by the responsible evaluation personnel, who prepare a summary of findings and recommendations for future surveillance or repair, including an analysis of reasons or suspected reasons for failure. Data derived from the inspections shall be reviewed by the Design Engineer so that the coatings condition assessment can be used to trend material condition and to provide strategic planning for required maintenance activities.

18.2.2 Plant-Specific Aging Management Programs

18.2.2.1 345 kV SF_6 Bus

The 345kV SF_6 Bus Program manages the aging that could lead to loss of pressure boundary due to elastomer degradation, loss of material due to corrosion and loss of function due to unacceptable air, moisture or SO_2 levels. Sulfur Hexafluoride (SF_6) is an inert gas which is used to insulate the bus conductor.

The program inspects for corrosion on the exterior of the bus duct housing, tests for leaks and tests gas samples to determine air, moisture and SO_2 levels.

The presence of air or moisture may lead to the loss of intended function. SO_2 levels are an indication of partial discharge internal to the bus.

The tests and inspections shall be performed prior to entering the period of extended of operation and at least once every six months thereafter.

18.2.2.2 Boral Monitoring

The Boral Monitoring Program assures the Boral neutron absorbers in the spent fuel racks maintain the validity of the criticality analysis in support of the rack design. The program relies on representative coupon samples mounted in a coupon "train" located in the spent fuel pool to monitor performance of the absorber material without disrupting the integrity of the storage system. Coupon samples are removed from the spent fuel pool on a prescribed schedule and physical, chemical and neutronic absorptive properties are measured. From these data, the physical condition and neutron-absorbing capacity of the Boral in the storage cells are assessed.

18.2.2.3 Nickel-Alloy Nozzle and Penetrations

The Nickel-Alloy Nozzles and Penetrations Program manages the aging effect of cracking due to primary water stress corrosion cracking (PWSCC) of nickel-alloy pressure boundary and structural components exposed to primary coolant.

The Nickel-Alloy Nozzles and Penetrations Program ranked the Alloy 600/82/182 locations based on four main criteria: PWSCC susceptibility (e.g., operational time and temperature), failure consequence, leakage detection margin, and radiation dose rates. Additionally, material heat susceptibility and other industry experience were considered.

The program incorporates the inspection schedules and frequencies for the nickel-alloy components in accordance with the plant ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program and, where applicable, ASME Code Case N-722 "*Additional Examinations for PWR [pressurized water reactor (PWR)] Pressure Retaining Welds in Class 1 Components Fabricated with Alloy 600/82/182 Materials, Section XI, Division 1*," subject to the conditions specified in 10 CFR 50.55a(g)(6)(ii)(E).

18.2.3 NUREG-1801 Chapter X Aging Management Programs

18.2.3.1 Metal Fatigue of Reactor Pressure Boundary

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 verify that the Cumulative Usage Fatigue (CUF) for reactor coolant system components remain less than 1.0 through the period of extended operation. The program determines the number of transients that occur and updates 60-year projections on a periodic basis.

The program is credited with monitoring reactor coolant system design transients. Cumulative Usage Fatigue of 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 program will use fatigue monitoring software to monitor the number of cycles a system or components endure. Pre-established limits will identify components approaching design limits. Components approaching design limits will be reanalyzed, repaired, replaced or inspected in accordance with applicable design codes.

18.2.3.2 Environmental Qualification (EQ) of Electric Components

The Environmental Qualification (EQ) of Electric Components Program provides a summary of the components that are managed for EQ aging. The program meets the requirements of 10 CFR 50.49 for the applicable electrical components important to safety. 10 CFR 50.49 defines the scope of components to be included, requires the preparation and maintenance of a list of in-scope components, and requires the preparation and maintenance of a qualification file that includes component performance specifications, electrical characteristics and the environmental conditions to which the components could be subjected. 10 CFR 50.49(e)(5) contains provisions for aging that require, in part, consideration of all significant types of aging degradation that can affect component functional capability. 10 CFR 50.49(e)(5) also requires replacement or refurbishment of components not qualified for the current license term prior to the end of designated life, unless additional life is established through ongoing qualification. Supplemental EQ regulatory guidance for compliance with these different qualification criteria is provided in NUREG-0588, "*Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment*," July 1981, and RG 1.89, Rev. 1, "*Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants, June 1984"*. Seabrook Station conforms to NUREG-0588 Category 1 requirements.

Seabrook Station's compliance with 10 CFR 50.49 ensures that the component can perform its intended functions during accident conditions after experiencing the effects of in-service aging. The EQ Program manages component thermal, radiation and cyclical aging, as applicable, through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods. As required by 10 CFR 50.49, EQ components not qualified for the current license term are to be refurbished, replaced, or have their qualification extended prior to reaching the aging limits established in the evaluation. Aging evaluations for electrical components in the EQ Program that specify a qualification of at least 40 years are TLAAs for license renewal because the criteria contained in 10 CFR 54.3 are met.

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 disposition option 10 CFR 54.2(c)(1)(iii), which states that the effects of aging will be adequately managed for the period of extended operation, is chosen and the EQ Program will manage the aging effects of the components**.**

18.2.4 Time-Limited Aging Analyses

18.2.4.1 Neutron Embrittlement of the Reactor Vessel

The current license period reactor vessel embrittlement analyses that evaluate reduction of fracture toughness of the Seabrook Station reactor vessel beltline materials are based on a 40-year End-of-Life (EOL) fluence values. The analyses associated with neutron embrittlement of reactor vessel materials due to neutron irradiation are Time-Limited Aging Analyses (TLAAs) as defined by 10 CFR 54.21(c) and must be evaluated for the increased neutron fluence associated with 60 years of operation.

The following Seabrook Station analyses are TLAAs that address the effects of neutron irradiation on the reactor vessel.

- Neutron Fluence
- Upper-Shelf Energy (USE)

- Pressurized Thermal Shock
- Pressure-Temperature (P-T) Limits

18.2.4.1.1 Neutron Fluence Analyses

The neutron fluence analysis is a TLAA as defined by 10 CFR 54.21(c) and must be evaluated for the increased neutron fluence associated with 60 years of operation. These neutron fluence projections are used as input to the analyses for fracture toughness, or Upper Shelf Energy (USE), Pressurized Thermal Shock (PTS) limits, Reference Temperatures – Nil Ductility Transition (RT_{NDT}), Adjusted Reference Temperatures (ART), Low-Temperature Overpressure Protection (LTOP) limits, and Reactor Vessel Pressure-Temperature Limit (P-T limit) curves.

18.2.4.1.2 Upper Shelf Analyses

The current Charpy Upper Shelf Energy (USE) analyses were prepared for the reactor vessel beltline materials for Seabrook Station based upon projected neutron fluence values for 40 years of service. These are TLAAs requiring evaluation using the projected 60-year fluence values.

The Seabrook Station analyses have been projected to the end of the period of extended operation for reactor vessel materials with projected fluence exceeding 1×10^{17} n/cm^2 (MeV > 1.0). 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 for Seabrook Station in accordance with 10 CFR 54.21(c)(1)(ii).

18.2.4.1.3 Pressurized Thermal Shock

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. The current RT_{PTS} analyses, evaluated for 32 Effective Full Power Years (EFPY) fluence values predicted for 40 years of operation, are TLAAs requiring evaluation for 60 years.

The margin is the difference between the maximum nil-ductility reference temperature (RT_{PTS}) 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 300ºF for beltline circumferential weld materials. If the calculated value reference temperature is less than the specified screening criterion, then the vessel is acceptable with respect to reactor vessel during postulated transients during the period of extended operation.

The RT_{PTS} analyses have been projected to the end of the period of extended operation and are shown to be within the maximum allowable PTS screening criteria limits in accordance with 10 CFR 54.21(c)(1)(ii).

18.2.4.1.4 Reactor Vessel Pressure-Temperature Limits, Including Low Temperature Overpressure Protection Limits

Title 10 CFR Part 50, Appendix G requires that the reactor pressure vessel be maintained within established pressure-temperature (P-T) limits, including heatup and cooldown operations. The P-T limits must account for the anticipated reactor vessel fluence. The current Low Temperature Overpressure Protection (LTOP) system uses a combination of residual heat removal suction relief valves and/or power operated relief valves as identified in Technical Specifications.

The current Seabrook Station P-T Low Temperature Overpressure Protection (LTOP) limit calculations are effective through 55 EFPY. The 55 EFPY P-T curves and LTOP limits meet the criteria of ASME Code, Section XI, Appendix G, and are in compliance with the fracture toughness requirements of 10 CFR 50.60 and 10 CFR 50, Appendix G through 55 EFPY.

The effects of aging on the intended function(s) will be adequately managed for the period of extended operation in accordance with 10 CFR 54(c)(1)(iii).

18.2.4.2 Metal Fatigue of Vessels and Piping

Metal fatigue was evaluated in the design process for Seabrook Station 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. This section is divided into seven subsections that each addresses a specific grouping of components that were analyzed in accordance with the same design requirements.

These groupings are as follows:

- Nuclear Steam Supply System (NSSS) Pressure Vessel and Component Fatigue Analyses
- Supplementary ASME Section III, Class 1 Piping and Component Fatigue Analyses
- Reactor Vessel Internals Fatigue Analyses
- Environmentally-Assisted Fatigue Analyses
- Steam Generator Tube, Loss of Material and Fatigue Usage from Flow-induced Vibration
- Absence of TLAAs For Fatigue Crack Growth, Fracture Mechanics Stability, or Corrosion Analyses Supporting Repair of Alloy 600 Materials
- Non-class I Component Failure Analysis

18.2.4.2.1 Nuclear Steam Supply System (NSSS) Pressure Vessel and Component Fatigue Analyses

Nuclear Steam Supply System (NSSS) pressure vessels and components for Seabrook Station were designed in accordance with ASME Section III, Class 1 requirements and were required to have explicit analyses of cumulative usage fatigue. The major components are the Reactor Vessel, Vessel Closure Head, Steam Generators and the Reactor Coolant Pump Casings. The applicable design codes for these components have been identified.

In order to determine if the ASME Section III, Class 1 fatigue analyses will remain valid for 60 years of service, a review of fatigue monitoring data was performed to determine the number of cumulative cycles of each transient type that have occurred during past plant operations. Then the average rate of occurrence was determined, and predictions of future transient occurrences were made. For each transient type, the 60-year projected number of occurrences was determined by adding the number of past occurrences to the number of predicted future occurrences. These 60-year projections were then compared to the numbers of design cycles used in the fatigue analyses to determine if the design cycles remain bounding for 60 years of operations. If the 60-year projected numbers of cycles is less than the numbers of cycles used in the design fatigue analyses, then the fatigue analyses based upon the design transients will remain valid for 60 years of operation if the design transient severity is also bounding of the actual transient severity.

The 40-year design transients bound the numbers of cycles projected to occur during 60 years of plant operations at Seabrook Station. Therefore, the NSSS Class 1 fatigue analyses that are based upon the 40-year design transients remain valid for the period of extended operation.

18.2.4.2.2 Supplementary ASME Section III, Class 1 Piping and Component Fatigue Analyses

In addition to the original design assumptions, the Seabrook Station Pressurizer fatigue evaluations were updated to include the added thermal stratification effects of insurge and outsurge events on the pressurizer lower head and surge nozzle.

Each of the Seabrook Station piping systems, including the Reactor Coolant System main loop piping, were originally designed in accordance with ASME Section III 1971 Edition with addenda through Winter 1972. 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. These transients included thermal stratification of the pressurizer surge line, as described in NRC Bulletin 88-11.

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.

18.2.4.2.2.1 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 (1990) and determined that the surge line will remain within the ASME Code requirements for the design life of the unit. The controlling fatigue location was the surge line hot leg nozzle safe-end. In later evaluations, plant-specific ASME Section III, Class 1 evaluations were performed for the surge line hot leg nozzle and pressurizer surge nozzle.

The hot leg surge line nozzle was evaluated for the effects of pressurizer insurge and outsurge transients and surge line stratification. Projected 60-year cycles of surge line stratification and insurge and outsurge transients were used when these were greater than previously evaluated design cycles. These evaluations resulted in CUF less than 1.0 at the hot let surge line nozzle.

The pressurizer surge nozzle was evaluated for the effects of pressurizer insurge and outsurge transients and surge line stratification. The pressurizer surge nozzle has been evaluated using an ASME Section III, Class 1 fatigue analysis. This analysis was part of the evaluation of the structural weld overlay applied to the pressurizer surge nozzle. This evaluation resulted in a Cumulative Usage Factor (CUF) less than 1.0 at the pressurizer surge nozzle.

The analyses remain valid for the period of extended operation for the Pressurizer Surge Line, Pressurizer Surge Nozzle and Surge Line Hot Leg Nozzle in accordance with 10 CFR 54.21(c)(1)(i).

18.2.4.2.2.2 Reactor Vessel Internals Aging Management

The Seabrook Station Reactor Vessel Internals were designed and constructed prior to the development of ASME Code requirements for core support structures. Demonstration that the effects of aging degradation are adequately managed is essential for assuring continued functionality of the reactor internals during the desired plant operating period, including license renewal. The Metal Fatigue of Reactor Coolant Pressure Boundary Program will monitor the number of design cycles assumed in the fatigue analysis to assure that these will not be exceeded during the period of extended operation.

In accordance with 10 CFR 54.21(c), the Metal Fatigue of Reactor Coolant Pressure Boundary Program will monitor the number of design cycles assumed in the fatigue analysis to the Aging

Management Program for reactor vessel internals will provide assurance that the effects of aging will be adequately managed for the period of extended operation per 10 CFR 54.21(c)(1)(iii).

18.2.4.2.3 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 these 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 a newer 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
4. Charging Nozzle
5. Safety Injection Nozzle
6. Residual Heat Removal System Class 1 Piping

For the NUREG/CR-6260 locations identified above, the plant-specific components were identified and the design ASME fatigue usage factors were adjusted by the environmentally-assisted fatigue penalty factors (F_{en}) to obtain the environmentally-assisted fatigue (EAF) result.

All locations were shown to achieve air-curve cumulative usage factors less than 1.0 for the 60 years of service. The evaluation of environmental fatigue effects for the Reactor Vessel Shell and Lower Head and Reactor Vessel Inlet and Outlet Nozzles determined that the CUF will remain below the ASME code allowable fatigue limit of 1.0 using the maximum applicable F_{en} , applied to CUF based on the design number of transients for these locations, when extended to 60 years. The remainder of these locations, Reactor Coolant System (RCS) Pressurizer Surge Line Nozzle, RCS Charging Nozzle, RCS Safety Injection Nozzle, and RCS Residual Heat Removal System Class 1 Piping, were analyzed in accordance with ASME Code Section III, Sub article NB-3200 using all six stress components. The evaluations show that EAFs exceed 1.0 for 60-years of service for the hot leg surge line nozzle and charging nozzle. These analyses were based on Seabrook Station specific conditions and these locations will be monitored for fatigue usage including environmental effects by the Metal Fatigue of Reactor Coolant Pressure Boundary Program. Specifically, this program will monitor critical transients to verify cycle limits are maintained below limits specified in the UFSAR. Pre-established action limits will permit completion of corrective actions before the design basis number of events is exceeded, and before the cumulative usage factor, including environmental effects, exceeds the ASME Code limit of 1.0. At least 2 years prior to entering the period of extended operation, Seabrook Station will implement the following aging management program for the plant-specific locations listed in NUREG/CR-6260 for the newer vintage Westinghouse plants.

1. Consistent with the Metal Fatigue of Reactor Coolant Pressure Boundary Program, Seabrook Station will update the fatigue usage calculations using refined fatigue analyses, if necessary, to determine acceptable CUFs (i.e., less than 1.0) when accounting for the effects of the reactor water environment. This includes applying the appropriate F_{en} factors to valid CUFs determined from an existing fatigue analysis valid for the period of extended operation or from an analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case). Formulas for calculating the environmental life correction factors for carbon and low alloy steels are contained in NUREG/CR-6583 and those for austenitic stainless steels are contained in NUREG/CR-5704. NUREG/CR-6909 includes alternate formulas for calculating environmental life correction factors, in addition to updated fatigue design curves.

2. If acceptable CUFs cannot be demonstrated for all the selected locations, then additional plant-specific locations will be evaluated. For the additional plant-specific locations, if CUF, including environmental effects is greater than 1.0, then Corrective Actions will be initiated, in accordance with the Metal Fatigue of Reactor Coolant Pressure Boundary Program.

   Therefore, the effects of the reactor coolant environment on fatigue usage factors in the remaining locations will be managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

18.2.4.2.4 Steam Generator Tube Fatigue Usage from Flow-Induced Vibration

The Seabrook Station Model F steam generators were evaluated with respect to flow induced vibration for the power increases that were implemented as part of the Seabrook Station Power Uprates. The analysis of the effects of steam generator flow-induced vibration fatigue usage assumed 40 years of operation.

Low-cycle fatigue usage for the most limiting tube in the most limiting power-uprated operating condition resulting from the flow-induced vibration tube bending stress is 0.2 ksi. This value is well below the fatigue endurance limit of 20 ksi at 1E+11 cycles, resulting in a computed fatigue usage of 0.0. High-cycle fatigue usage of U-bend tubes was evaluated. One of the prerequisites for high-cycle U-bend fatigue is a dented support condition at the upper plate. Seabrook Station steam generator tube support plates are manufactured from stainless steel therefore there is no potential for the necessary conditions to occur. It was concluded that the support condition leading to a dented support condition necessary for high-cycle fatigue cannot occur in the Seabrook Station Model F steam generators.

18.2.4.2.5 Non-Class 1 Component Fatigue Analyses

This section describes fatigue-related TLAAs arising within design analyses of the Non-Class 1 piping and components. These piping and tubing components can be designed in accordance with ASME Section III Class 2 and 3.

The following non-Class 1 Seabrook Station systems that are in scope for license renewal were designed in accordance with ASME Section III Class 2 and 3, requirements: Reactor Coolant System (including primary loop piping and pressurizer surge line piping), Chemical and Volume Control System, Safety Injection System, Primary Component Cooling Water, Service Water, Sample System, Residual Heat Removal System, Main Steam System, Main Condensate and Feedwater, and the Steam Generator Blowdown System.

In order to evaluate these TLAAs for 60 years, the number 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 maximum number of cycles that would not result in reduction of the allowable stress range, 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 projection results for Seabrook Station 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 the systems designed to ASME Section III Class 2 and 3, requirements.

The Sample System thermal cycles do not trend along with operational cycles because sampling is required on a periodic basis, as opposed to an operational basis. However, only the portion of the sampling lines that constitutes piping need be considered here. In this case that is a very short section of piping directly connected to the Reactor Coolant System (RCS) loop piping. Since this section of piping has no isolation valve and no bends, it is assumed to always be exposed to primary loop temperature and pressure condition. Similarly, since there are no other external piping connections (only the tubing connection exits), the line will not experience any other externally applied loads. Therefore, that section of the sampling line that constitutes ASME Section III Class 2 and will only experience the RCS loop transients which have already been shown to be less than 7,000 cycles and the line is, therefore, acceptable.

As shown in UFSAR Table 3.2-2 there are several sections of ANS Safety Class NNS (Non-Nuclear Safety) piping which the principal design code is B31.1 and are seismic Category I. These piping, piping components or piping elements are within the scope of license renewal for a(2) as a failure could affect an a(1) classified component.

The 60-year transient projection results for Seabrook Station 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 the 7,000 cycles limit requiring reduction of the allowable thermal moment range in ASME Section III Class 2 and 3 and B31.1 rules. Therefore, there is no impact upon the implicit fatigue analyses used in the component design for the systems designed to ASME Section III Class 2 and 3 requirements. The same argument applies to the cyclic thermal cycles on the non-nuclear safety classified components (including B31.1) of these systems that are within the scope of license renewal.

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

18.2.4.3 Environmental Qualification (EQ) of Electric Components

In accordance with 10CFR50.49, all electrical equipment important to safety located in a harsh environment and required to function in that environment must be environmentally qualified. In order for a component to have sufficient design margin to perform its important to safety function under harsh environment conditions, the component may need to be periodically rebuilt or replaced. For these EQ components, the EQ program insures that they are rebuilt, replaced or reevaluated at the necessary interval. All qualified lives within the scope of the EQ program are managed under the EQ Program.

The Seabrook Station Environmental Qualification (EQ) of Electric Components Program implements aging management activities which are credited for the management of aging in selected components within the scope of 10 CFR 54.

18.2.4.4 Fatigue of the Containment Liner and Penetrations

The original design analysis for the Seabrook Station containment liner plate determined that all of the criteria specified in ASME Section III Article NE-3221.5(d) required for exemption from the requirement to perform a cyclic operation analysis were met. To address these 40-year cycles during the period of extended operation, a re-evaluation of the six fatigue exemption requirements utilizing anticipated 60-year stress cycles was performed. The result of this analysis determined that the specified conditions through the period of extended operation continue to satisfy the requirement for exemption from analysis for cyclic operation in accordance with in ASME Section III Article NE-3221.5(d). The analysis has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

Specific cyclic evaluations are listed in the Seabrook Station UFSAR Section 3.8.2.3 for the Personnel Airlock, Equipment Hatch and Fuel Transfer Tube therefore TLAAs are considered. The analyses for the Personnel Airlock, Equipment Hatch and Fuel Transfer Tube remains valid for the period of extended operation as the anticipated number of cycles anticipated during the period of extended operation is bounded by the original design in accordance with 10 CFR 54.21(c)(1)(i).

18.2.4.5 Other Plant-Specific TLAAs

18.2.4.5.1 Reactor Coolant Pump Flywheel Fatigue Crack Growth Analyses

Westinghouse Report WCAP-14535-A, Rev. 0, "*Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination*" 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 reactor coolant pump (RCP) flywheel inservice inspection requirements for all operating Westinghouse plants and certain Babcock and Wilcox plants. The number of cycles (pump starts and stops) used in this report was 6,000 for a 60-year plant life. Crack growth was shown to be negligible from exposure to these 6,000 cycles.

Based on WCAP 15666; Amendment 134 to the Facility Operating License extended the reactor coolant pump (RCP) flywheel examination frequency from a 10-year inspection interval to an interval not to exceed 20 years. During the period of extended operation, the reactor coolant pump flywheels will be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision I, August 1975. In lieu of Position C.4.b(I) and C.4.b(2), this inspection shall be by either of the following examinations:

a. An in-place examination, utilizing ultrasonic testing, over the volume from the inner bore of the flywheel to the circle of one-half the outer radius; or

b. A surface examination, utilizing magnetic particle testing and/ or penetrant testing, of the exposed surfaces of the disassembled flywheel.

Based on the current cycle count projected to 60 years, the projected cycle count is much less than the analyzed cycle counts of 6,000 cycles. The reactor coolant pump flywheel analysis remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

18.2.4.5.2 Leak-Before-Break Analyses

Title 10 Code of Federal Regulations Part 50 Appendix A, Criterion 4 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.

Based on loading, pipe geometry, and fracture toughness considerations, enveloping governing locations were determined at which LBB crack stability evaluations were made. Through-wall flaw sizes were found which would cause a leak at a rate of ten (10) times the leakage detection system capability of the plant. Large margins for such flaw sizes were demonstrated against flaw instability. Finally, fatigue crack growth was shown not to be an issue for the reactor coolant system primary loop piping. The thermal transients used in the fatigue crack growth analysis were the design transients listed in the NSSS Design Limits for 40 years at Seabrook Station. The corresponding 60-year projected cycles are lower than the 40-year design values. Therefore, the numbers of design cycles assumed in the analysis bound the numbers of design cycles projected for the period of extended operation.

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

18.2.4.5.3 High Energy Line Break Postulation Based on Cumulative Usage Factor

The Seabrook Station High Energy Line Break (HELB) analysis used a screening criterion of CUF greater than 0.1 to identify areas of investigation. The Seabrook Station Updated Final Safety Analysis Report (UFSAR) Section 3.6(B).2.1(a) provided a basis to eliminate locations in each piping run or branch run from further consideration as high energy line break locations on the basis of low fatigue including intermediate location when the CUF was less than 0.1.

Selection of pipe failure locations for evaluation of the consequences on nearby essential systems, components, and structures, except for the reactor coolant loop, is in accordance with Regulatory Guide 1.46, and NRC Branch Technical Positions ASB 3-1 and MEB 3-1. A revised stress analysis also permitted omission of the surge line intermediate breaks. A leak-before-break (LBB) analysis eliminated large breaks in the main reactor coolant loops.

The surge line intermediate break locations were eliminated based on usage factor. The most recent piping analysis confirmed the elimination of these break locations. The analysis that justified the elimination of these intermediate locations in the surge line is therefore a TLAA.

Since the 60 year projected cycles are bounded by the original design cycles, the present intermediate locations with CUF less than 0.1 remain valid for the period of extended operation.

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

18.2.4.5.4 Fuel Transfer Tube Bellows Design Cycles

The fuel transfer tube assembly connects the fuel transfer canal (inside the containment structure) to the transfer pool (inside the spent fuel handling building). The fuel transfer tube assembly passes through the containment wall and through the exterior wall of the spent fuel handling building. The fuel transfer tube assembly is comprised of a 24-inch diameter penetration sleeve penetrating through the containment and spent fuel 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.

The fatigue analysis for each of the three bellows is based on the consideration of 20 occurrences of the Operating Basis Earthquake, each occurrence having 20 cycles of maximum response therefore, this design analysis is a TLAA requiring evaluation for the period of extended operation.

It is projected that 1 OBE would occur for Seabrook Station in 60 years of operation. Since the number of occurrences projected for 60 years is below the design limit of 5 occurrences of 10 cycles, the design analysis remains valid for the period of extended operation.

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

18.2.4.5.5 Crane Load Cycle Limits

18.2.4.5.5.1 Polar Gantry Crane

The design specification for the 420/50-ton Polar Crane in the containment structure at Seabrook Station required that the crane conform to the design requirements of Crane Manufacturers Association of America (CMAA) Specification 70, "*Specifications for Electric Overhead Traveling Cranes*". Service requirements specified for the design of this crane correspond 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.

The estimated number of lifts for the Polar Crane over the remaining 40 years of service (which includes 20 years of Extended Operation) is 19,440 with most of the lifts being less than 2500 pounds. This rate is based on refueling outage use, therefore, the first 20 years of service life for the Polar Crane would include approximately 10,000 load cycles. Thus, the total service life load cycles will be approximately 30,000. Since the total number of lifts is less than the allowable design value of up to 100,000 cycles, the Polar Crane load cycle fatigue analyses for Seabrook Station remains valid for 60 years of plant operation.

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

18.2.4.5.5.2 Cask Handling Crane

The original Seabrook Station Cask Handling Crane was replaced in 2008 by a single failure-proof crane rated for 130 tons (main hoist) and 5 tons (each of two auxiliary hoists). To meet single failure criteria, each of these cranes was designed to the requirement of ASME NOG-1-2004, NUREG-0554, and NUREG-0612. The cranes were also designed to Crane Manufacturers Association of America (CMAA) Specification 70, "*Specifications for Electric Overhead Traveling Cranes*", with an allowable design life cycle range of up to 100,000 cycles. This evaluation of cycles over the projected 40-year life is the basis of a safety determination and has been identified as a TLAA requiring evaluation for the period of extended operation.

The projected number of major lifts for the Cask Handling Crane is less than 500 cycles. This estimate is based upon the expected number of casks that must be handled during each cask loading campaign and the projected number of campaigns through the period of extended operation. Allowing for double that number for minor lifts, or 1000 cycles, the estimated number of lifts for the Cask Handling Crane, 1500 cycles, is much less than the maximum allowable design value of 100,000 cycles, the Cask Handling Crane load cycle fatigue analyses remain valid for 60 years of plant operation.

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

18.2.4.5.6 Service Level I Coatings Qualification

Service Level 1 coatings used at Seabrook Station are in compliance with the applicable ANSI standards for coating systems inside containment. In a design basis accident, the Emergency Core Cooling System (ECCS) at Seabrook Station pumps water from inside the containment sump to the reactor vessel to keep the core covered with water and make up losses from the pipe break location. These coatings could potentially detach during a design basis accident and the coating debris could contribute to flow blockage of ECCS suction strainers. The ECCS has suction piping located below the waterline inside the sump. Since it is assumed that the degree of radiation exposure used in the original qualification testing was intended to bound 40 years of operation, qualification of Service Level 1 coatings is considered a TLAA.

Seabrook Station Service Level I Coatings are managed by the Protective Coating Monitoring and Maintenance Program and Procedure for Application of Service Level I Coatings. Seabrook Station periodically conducts condition assessments of Service Level I coatings inside containment.

The periodic condition assessments, and the resulting repair/replacement activities, assure that the amount of Service Level 1 coatings which may be susceptible to detachment from the substrate during a LOCA event is minimized. The program provides for maintenance of coatings for the period of extended operation in accordance with 10 CFR 54.21 (c)(1)(iii).

18.2.4.5.7 <u>Canopy Seal Pressure Housings</u>

The canopy seal clamp assemblies were designed for a 40-year design life on the basis of meeting stress limits. The original fatigue analysis considered the forces that would be applied to the center of the canopy Seal-Pressure Housings (a/k/a head adapter) which maximized the moments on the J-Grove weld and moment along the length of the adapter. The fatigue analysis for the Canopy Seal Pressure Housings is based on the consideration of 400 cycles consisting of 20 occurrences of the Operating Basis Earthquake, each occurrence having 20 cycles of maximum response. This design analysis is a TLAA requiring evaluation for the period of extended operation.

It is projected that 1 OBE would occur for Seabrook Station in 60-years of operation. Since the number of occurrences projected for 60-years is below the design limit of 5 occurrences of 10 cycles the design analysis remains valid for the period of extended operation.

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

18.2.4.5.8 <u>Hydrogen Analyzer</u>

The Seabrook Station Hydrogen Analyzer was evaluated with respect to radiation exposure. The UFSAR contains accumulated radiation dose limits for a 40-year operating period.

The operational dose for 40-year is 5 x 10^6 rads.

The projected maximum 40-year exposure comes from three sources; the gas in the analyzers themselves, the gas in the piping in the room, and the shine from the containment atmosphere through the penetrations into the room. The dose to the recombiner from these three sources is 7.2 x 10^3 rads annually. This leads to a projected 60-year dose of 4.32 x 10^5 rads which is less than the 40-year design dose.

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

18.2.4.5.9 <u>Mechanical Equipment Qualification</u>

The Seabrook Station CLB commits to the review and evaluation of the environmental qualification of mechanical equipment to demonstrate compliance with 10 CFR Part 50 General Design Criteria Appendix A.

Results of this evaluation demonstrate safety-related active mechanical equipment located in harsh environments had been adequately addressed.

Since a period of 40 years was used to determine the normal service radiation exposure to the equipment, mechanical equipment qualification (MEQ) is considered a TLAA.

The design basis event conditions during the period of extended operation will remain the same as those in the current license period which have been adjusted to account for previously approved power uprate conditions. Therefore, the design basis event parameters, including the temperature, pressure and time profiles, do not require further evaluation as TLAAs for license renewal.

The effects of aging on the intended function(s) of equipment included under Mechanical Equipment Qualification are bounded by existing equipment design limits, in accordance with 10 CFR 54.21(c)(1)(i), for the period of extended operation.

18.2.4.5.10 Diesel Generator Thermal Cycle Evaluation

The Emergency Diesel Generators provide Emergency Power to Buses 5 and 6. The Emergency Diesel Generators were analyzed for thermal cycling by the engine manufacturer for Environmental Qualification in accordance with IEEE-323. The manufacturer qualified the Diesel Generator for 5454 Full-Temperature Cycles for the forty year design life of the plant. Under current plant operating practices, the Emergency Diesel Generators are operated only occasionally during periodic surveillance and maintenance testing. Monthly testing over 60 years would contribute 720 cycles. Assuming an equal number of starts for maintenance and actual events an additional 1440 cycles could occur. These actual and potential cycles combined equal slightly more than 2160 cycles for the Emergency Diesel Generators. The projected 60 year cycles is much less than the design basis thermal cycling for 40 years.

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

18.2.4.5.11 Metal Corrosion Allowances and Corrosion Effects

The Seabrook Station licensing basis assumes a general corrosion and erosion rate of 3 mils is for the steam generator tube wall. The corrosion rate is based on a conservative weight loss rate of Inconel tubing in flowing 650°F primary side reactor coolant fluid. The weight loss, when equated to a thinning rate and projected over a 40-year design operating objective, with appropriate reduction after initial hours, is equivalent to 0.083 mils thinning. A linear projection of this thinning rate to a 60-year period is equivalent to 0.1245 mils thinning. This linear projection to 60 years is considered to be conservative because it includes in the base rate the higher rate during the initial hours. The assumed corrosion rate of 3 mils leaves a conservative 2.8755 mils for general corrosion thinning on the secondary side.

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

18.2.4.5.12 Steam Generator Tube Wall Wear from Flow-Induced Vibration

The maximum predicted tube wall wear for a 40-year operating life was 0.0032 inch for the pre-uprate conditions. As a result of the 56% increase in the tube wear rate as a result of the power uprate analysis to 3659 MW_{th}, the maximum 40-year tube wall wear is less than 0.0050 inch. The maximum 60-year tube wall wear is 0.0075 inch (~20% through-wall wear) based on a linear time projection. This amount of tube wall wear is less than the limit of acceptability of 40% of wall thickness and is deemed not to significantly affect tube integrity.

The evaluation showed that significant levels of tube vibration will not occur from either the fluid elastic or turbulent mechanisms above those associated with the pre-uprated condition, thus justifying the linear projection.

No.	PROGRAM or TOPIC	COMMITMENT	UFSAR LOCATION	SCHEDULE
1.	PWR Vessel Internals	Provide confirmation and acceptability of the implementation of MRP-227-A by addressing the plant-specific Applicant/Licensee Action Items outlined in section 4.2 of the NRC SER.	18.2.1.7	Complete
2.	Closed-Cycle Cooling Water	Enhance the program to include visual inspection for cracking, loss of material and fouling when the in-scope systems are opened for maintenance.	18.2.1.12	Prior to the period of extended operation.
3.	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems	Enhance the program to monitor general corrosion on the crane and trolley structural components and the effects of wear on the rails in the rail system.	18.2.1.13	Prior to the period of extended operation.
4.	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems	Enhance the program to list additional cranes for monitoring.	18.2.1.13	Prior to the period of extended operation.
5.	Compressed Air Monitoring	Enhance the program to include an annual air quality test requirement for the Diesel Generator compressed air sub system.	18.2.1.14	Prior to the period of extended operation.
6.	Fire Protection	Enhance the program to perform visual inspection of penetration seals by a fire protection qualified inspector.	18.2.1.15	Prior to the period of extended operation.
7.	Fire Protection	Enhance the program to add inspection requirements such as spalling, and loss of material caused by freeze-thaw, chemical attack, and reaction with aggregates by qualified inspector.	18.2.1.15	Prior to the period of extended operation.
8.	Fire Protection	Enhance the program to include the performance of visual inspection of fire-rated doors by a fire protection qualified inspector.	18.2.1.15	Prior to the period of extended operation.
9.	Fire Water System	Enhance the program to include NFPA 25 (2011 Edition) guidance for “where sprinklers have been in place for 50 years, they shall be replaced or representative samples from one or more sample areas shall be submitted to a recognized testing laboratory for field service testing”.	18.2.1.16	Prior to the period of extended operation.

No.	PROGRAM or TOPIC	COMMITMENT	UFSAR LOCATION	SCHEDULE
10.	Fire Water System	Enhance the program to include the performance of periodic flow testing of the fire water system in accordance with the guidance of NFPA 25 (2011 Edition).	18.2.1.16	Prior to the period of extended operation.
11.	Fire Water System	Enhance the program to include the performance of periodic visual or volumetric inspection of the internal surface of the fire protection system upon each entry to the system for routine or corrective maintenance to evaluate wall thickness and inner diameter of the fire protection piping ensuring that corrosion product buildup will not result in flow blockage due to fouling. Where surface irregularities are detected, follow-up volumetric examinations are performed. These inspections will be documented and trended to determine if a representative number of inspections have been performed prior to the period of extended operation. If a representative number of inspections have not been performed prior to the period of extended operation, focused inspections will be conducted. These inspections will commence during the ten year period prior to the period of extended operation and continue through the period of extended operation	18.2.1.16	Within ten years prior to the period of extended operation.
12.	Aboveground Steel Tanks	Enhance the program to include 1) In-scope outdoor tanks, except fire water storage tanks, constructed on soil or concrete, 2) Indoor large volume storage tanks (greater than 100,000 gallons) designed to near-atmospheric internal pressures, sit on concrete or soil, and exposed internally to water, 3) Visual, surface, and volumetric examinations of the outside and inside surfaces for managing the aging effects of loss of material and cracking, 4) External visual examinations to monitor degradation of the protective paint or coating, and 5) Inspection of sealant and caulking for degradation by performing visual and tactile examination (manual manipulation) consisting of pressing on the sealant or caulking to detect a reduction in the resiliency and pliability.	18.2.1.17	Within 10 years prior to the period of extended operation.
13.	Fire Water System	Enhance the program to perform exterior inspection of the fire water storage tanks annually for signs of degradation and include an ultrasonic inspection and evaluation of the internal bottom surface of the two Fire Protection Water Storage Tanks per the guidance provided in NFPA 25 (2011 Edition).	18.2.1.16	Within ten years prior to the period of extended operation.

No.	PROGRAM or TOPIC	COMMITMENT	UFSAR LOCATION	SCHEDULE
14.	Fuel Oil Chemistry	Enhance program to add requirements to 1) sample and analyze new fuel deliveries for biodiesel prior to offloading to the Auxiliary Boiler fuel oil storage tank and 2) periodically sample stored fuel in the Auxiliary Boiler fuel oil storage tank.	18.2.1.18	Prior to the period of extended operation.
15.	Fuel Oil Chemistry	Enhance the program to add requirements to check for the presence of water in the Auxiliary Boiler fuel oil storage tank at least once per quarter and to remove water as necessary.	18.2.1.18	Prior to the period of extended operation.
16.	Fuel Oil Chemistry	Enhance the program to require draining, cleaning and inspection of the diesel fire pump fuel oil day tanks on a frequency of at least once every ten years.	18.2.1.18	Prior to the period of extended operation.
17.	Fuel Oil Chemistry	Enhance the program to require ultrasonic thickness measurement of the tank bottom during the 10-year draining, cleaning and inspection of the Diesel Generator fuel oil storage tanks, Diesel Generator fuel oil day tanks, diesel fire pump fuel oil day tanks and auxiliary boiler fuel oil storage tank.	18.2.1.18	Prior to the period of extended operation.
18.	Reactor Vessel Surveillance	Enhance the program to specify that all pulled and tested capsules, unless discarded before August 31, 2000, are placed in storage.	18.2.1.19	Prior to the period of extended operation.
19.	Reactor Vessel Surveillance	Enhance the program to specify that if plant operations exceed the limitations or bounds defined by the Reactor Vessel Surveillance Program, such as operating at a lower cold leg temperature or higher fluence, the impact of plant operation changes on the extent of Reactor Vessel embrittlement will be evaluated and the NRC will be notified.	18.2.1.19	Prior to the period of extended operation.
20.	Reactor Vessel Surveillance	Enhance the program as necessary to ensure the appropriate withdrawal schedule for capsules remaining in the vessel such that one capsule will be withdrawn at an outage in which the capsule receives a neutron fluence that meets the schedule requirements of 10 CFR 50 Appendix H and ASTM E185-82 and that bounds the 60-year fluence, and the remaining capsule(s) will be removed from the vessel unless determined to provide meaningful metallurgical data.	18.2.1.19	Prior to the period of extended operation.

No.	PROGRAM or TOPIC	COMMITMENT	UFSAR LOCATION	SCHEDULE
21.	Reactor Vessel Surveillance	Enhance the program to ensure that any capsule removed, without the intent to test it, is stored in a manner which maintains it in a condition which would permit its future use, including during the period of extended operation.	18.2.1.19	Prior to the period of extended operation.
22.	One-Time Inspection	Implement the One Time Inspection Program.	18.2.1.20	Within ten years prior to the period of extended operation.
23.	Selective Leaching of Materials	Implement the Selective Leaching of Materials Program. The program will include a one-time inspection of selected components where selective leaching has not been identified and periodic inspections of selected components where selective leaching has been identified.	18.2.1.21	Within five years prior to the period of extended operation.
24.	Buried Piping And Tanks Inspection	Implement the Buried Piping And Tanks Inspection Program.	18.2.1.22	Within ten years prior to the period of extended operation
25.	One-Time Inspection of ASME Code Class 1 Small Bore-Piping	Implement the One-Time Inspection of ASME Code Class 1 Small Bore-Piping Program.	18.2.1.23	Within ten years prior to the period of extended operation.
26.	External Surfaces Monitoring	Enhance the program to specifically address the scope of the program, relevant degradation mechanisms and effects of interest, the refueling outage inspection frequency, the training requirements for inspectors and the required periodic reviews to determine program effectiveness.	18.2.1.24	Prior to the period of extended operation.
27.	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components	Implement the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program.	18.2.1.25	Prior to the period of extended operation.
28.	Lubricating Oil Analysis	Enhance the program to add required equipment, lube oil analysis required, sampling frequency, and periodic oil changes.	18.2.1.26	Prior to the period of extended operation.
29.	Lubricating Oil Analysis	Enhance the program to sample the oil for the Reactor Coolant pump oil collection tanks.	18.2.1.26	Prior to the period of extended operation.

No.	PROGRAM or TOPIC	COMMITMENT	UFSAR LOCATION	SCHEDULE
30.	Lubricating Oil Analysis	Enhance the program to require the performance of a one-time ultrasonic thickness measurement of the lower portion of the Reactor Coolant pump oil collection tanks prior to the period of extended operation.	18.2.1.26	Prior to the period of extended operation.
31.	ASME Section XI, Subsection IWL	Enhance procedure to include the definition of “Responsible Engineer”.	18.2.1.28	Prior to the period of extended operation.
32.	Structures Monitoring Program	Enhance procedure to add the aging effects, additional locations, inspection frequency and ultrasonic test requirements.	18.2.1.31	Prior to the period of extended operation.
33.	Structures Monitoring Program	Enhance procedure to include inspection of opportunity when planning excavation work that would expose inaccessible concrete.	18.2.1.31	Prior to the period of extended operation.
34.	Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Implement the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program.	18.2.1.32	Prior to the period of extended operation.
35.	Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits	Implement the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits program.	18.2.1.33	Prior to the period of extended operation.
36.	Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Implement the Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program.	18.2.1.34	Prior to the period of extended operation.
37.	Metal Enclosed Bus	Implement the Metal Enclosed Bus program.	18.2.1.35	Prior to the period of extended operation.
38.	Fuse Holders	Implement the Fuse Holders program.	18.2.1.36	Prior to the period of extended operation.

No.	PROGRAM or TOPIC	COMMITMENT	UFSAR LOCATION	SCHEDULE
39.	Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Implement the Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program.	18.2.1.37	Prior to the period of extended operation.
40.	345 KV SF6 Bus	Implement the 345 KV SF6 Bus program.	18.2.2.1	Prior to the period of extended operation.
41.	Metal Fatigue of Reactor Coolant Pressure Boundary	Enhance the program to include additional transients beyond those defined in the Technical Specifications and UFSAR.	18.2.3.1	Prior to the period of extended operation.
42.	Metal Fatigue of Reactor Coolant Pressure Boundary	Enhance the program to implement a software program, to count transients to monitor cumulative usage on selected components.	18.2.3.1	Prior to the period of extended operation.
43.	Pressure –Temperature Limits, including Low Temperature Overpressure Protection Limits	Seabrook Station will submit updates to the P-T curves and LTOP limits to the NRC at the appropriate time to comply with 10 CFR 50 Appendix G.	18.2.4.1.4	Complete

No.	PROGRAM or TOPIC	COMMITMENT	UFSAR LOCATION	SCHEDULE
44.	Environmentally-Assisted Fatigue Analyses (TLAA)	NextEra Seabrook will perform a review of design basis ASME Class 1 component fatigue evaluations to determine whether the NUREG/CR-6260-based components that have been evaluated for the effects of the reactor coolant environment on fatigue usage are the limiting components for the Seabrook plant configuration. If more limiting components are identified, the most limiting component will be evaluated for the effects of the reactor coolant environment on fatigue usage. If the limiting location identified consists of nickel alloy, the environmentally-assisted fatigue calculation for nickel alloy will be performed using the rules of NUREG/CR-6909. (1) Consistent with the Metal Fatigue of Reactor Coolant Pressure Boundary Program Seabrook Station will update the fatigue usage calculations using refined fatigue analyses, if necessary, to determine acceptable CUFs (i.e., less than 1.0) when accounting for the effects of the reactor water environment. This includes applying the appropriate Fen factors to valid CUFs determined from an existing fatigue analysis valid for the period of extended operation or from an analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case). (2) If acceptable CUFs cannot be demonstrated for all the selected locations, then additional plant-specific locations will be evaluated. For the additional plant-specific locations, if CUF, including environmental effects is greater than 1.0, then Corrective Actions will be initiated, in accordance with the Metal Fatigue of Reactor Coolant Pressure Boundary Program, B.2.3.1. Corrective Actions will include inspection, repair, or replacement of the affected locations before exceeding a CUF of 1.0 or the effects of fatigue will be managed by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method accepted by the NRC).	18.2.4.2.3	At least two years prior to the period of extended operation.

No.	PROGRAM or TOPIC	COMMITMENT	UFSAR LOCATION	SCHEDULE
45.	Alkali-Silica Reaction (ASR) Monitoring Program	NextEra will obtain additional cores in the vicinity of 20% of the extensometers and perform modulus testing. Using these test results, NextEra will determine the change in through-thickness expansion since installation of the extensometers and compare it to change determined from extensometer readings. Consistency between these results will provide additional corroboration of the methodology in MPR-4153.	18.2.1.31.A	At least 5 years prior to the period of extended operation (initial study) and 10 years thereafter (follow-up study).
46.	Protective Coating Monitoring and Maintenance	Enhance the program by designating and qualifying an Inspector Coordinator and an Inspection Results Evaluator.	18.2.1.38	Prior to the period of extended operation.
47.	Protective Coating Monitoring and Maintenance	Enhance the program by including, "Instruments and Equipment needed for inspection may include, but not be limited to, flashlight, spotlights, marker pen, mirror, measuring tape, magnifier, binoculars, camera with or without wide angle lens, and self sealing polyethylene sample bags."	18.2.1.38	Prior to the period of extended operation.
48.	Protective Coating Monitoring and Maintenance	Enhance the program to include a review of the previous two monitoring reports.	18.2.1.38	Prior to the period of extended operation.
49.	Protective Coating Monitoring and Maintenance	Enhance the program to require that the inspection report is to be evaluated by the responsible evaluation personnel, who is to prepare a summary of findings and recommendations for future surveillance or repair.	18.2.1.38	Prior to the period of extended operation.
50.	ASME Section XI, Subsection IWE	Perform UT of the accessible areas of the containment liner plate in the vicinity of the moisture barrier for loss of material. Perform opportunistic UT of inaccessible areas.	18.2.1.27	Baseline inspections were completed during OR16. Repeat containment liner UT thickness examinations at intervals of no more than five (5) refueling outages.

No.	PROGRAM or TOPIC	COMMITMENT	UFSAR LOCATION	SCHEDULE
51.	Bolting Integrity	Enhance the program to manage the aging effects for closure bolting within air and gas filled systems by using an applicable inspection technique that ensures the integrity of bolted joints will be demonstrated. For closure bolting within systems at atmospheric pressure, tightness checks will be performed on 20 percent of bolts with a maximum of 25 bolts per population. Populations will be of the same material and environment combination. Inspections will occur before the period of extended operation, and then every 10 years after the initial inspection date.	18.2.1.9	Prior to the period of extended operation.
52.	ASME Section XI, Subsection IWL	Implement measures to maintain the exterior surface of the Containment Structure, from elevation -30 feet to +20 feet, in a dewatered state.	18.2.1.28	Complete
53.	Reactor Head Closure Studs	Replace the spare reactor head closure stud(s) manufactured from the bar that has a yield strength > 150 ksi with ones that do not exceed 150 ksi.	18.2.1.3	Prior to the period of extended operation.
54.	Steam Generator Tube Integrity	NextEra will address the potential for cracking of the primary to secondary pressure boundary due to PWSCC of tube-to-tubesheet welds using one of the following two options: 1) Perform a one-time inspection of a representative sample of tube-to-tubesheet welds in all steam generators to determine if PWSCC cracking is present and, if cracking is identified, resolve the condition through engineering evaluation justifying continued operation or repair the condition, as appropriate, and establish an ongoing monitoring program to perform routine tube-to-tubesheet weld inspections for the remaining life of the steam generators, or 2) Perform an analytical evaluation showing that the structural integrity of the steam generator tube-to-tubesheet interface is adequately maintaining the pressure boundary in the presence of tube-to-tubesheet weld cracking, or redefining the pressure boundary in which the tube-to-tubesheet weld is no longer included and, therefore, is not required for reactor coolant pressure boundary function. The redefinition of the reactor coolant pressure boundary must be approved by the NRC as part of a license amendment request.	18.2.1.10	Complete

No.	PROGRAM or TOPIC	COMMITMENT	UFSAR LOCATION	SCHEDULE
55.	ASME Section XI, Subsection IWL	Concrete Surface Suspect areas identified during the 2010 and 2016 Containment IWL inspections will be incorporated into the Seabrook Station Containment Inservice Inspection (CISI) Plan.	18.2.1.28	September 1, 2020
56.	Closed-Cycle Cooling Water System	Revise the station program documents to reflect the EPRI Guideline operating ranges and Action Level values for hydrazine and sulfates.	18.2.1.12	Prior to the period of extended operation.
57.	Closed-Cycle Cooling Water System	Revise the station program documents to reflect the EPRI Guideline operating ranges and Action Level values for Diesel Generator Cooling Water Jacket pH.	18.2.1.12	Prior to the period of extended operation.
58.	Fuel Oil Chemistry	Update Technical Requirement Program 5.1, (Diesel Fuel Oil Testing Program) ASTM standards to ASTM D2709-96 and ASTM D4057-95 required by the GALL XI.M30 Rev 1	18.2.1.18	Prior to the period of extended operation.
59.	Nickel Alloy Nozzles and Penetrations	The Nickel Alloy Aging Nozzles and Penetrations program will implement applicable Bulletins, Generic Letters, and staff accepted industry guidelines.	18.2.2.3	Prior to the period of extended operation.
60.	Buried Piping and Tanks Inspection	Implement the design change replacing the buried Auxiliary Boiler supply piping with a pipe-within-pipe configuration with leak detection capability.	18.2.1.22	Prior to the period of extended operation.
61.	Compressed Air Monitoring Program	Replace the flexible hoses associated with the Diesel Generator air compressors on a frequency of every 10 years.	18.2.1.14	Within ten years prior to the period of extended operation.
62.	Water Chemistry	Enhance the program to include a statement that sampling frequencies are increased when chemistry action levels are exceeded.	18.2.1.2	Prior to the period of extended operation.
63.	Flow Induced Erosion	Ensure that the quarterly CVCS Charging Pump testing is continued during the PEO. Additionally, add a precaution to the test procedure to state that an increase in the CVCS Charging Pump mini flow above the acceptance criteria may be indicative of erosion of the mini flow orifice as described in LER 50-275/94-023.	18.2.1.2	Prior to the period of extended operation.

No.	PROGRAM or TOPIC	COMMITMENT	UFSAR LOCATION	SCHEDULE
64.	Buried Piping and Tanks Inspection	Soil analysis shall be performed prior to entering the period of extended operation to determine the corrosivity of the soil in the vicinity of non-cathodically protected steel pipe within the scope of this program. If the initial analysis shows the soil to be non-corrosive, this analysis will be re-performed every ten years thereafter.	18.2.1.22	Within ten years prior to the period of extended operation.
65.	Flux Thimble Tube	Implement measures to ensure that the movable incore detectors are not returned to service during the period of extended operation.	N/A	Complete.
66.	Alkali-Silica Reaction (ASR) Monitoring Program	NextEra will perform an integrated review of expansion trends at Seabrook Station by conducting a periodic assessment of ASR expansion behavior to confirm that the MPR/FSEL large-scale test programs remain applicable to plant structures. This review will include the following specific considerations: Review of all cores removed to date for trends of any indications of mid-plane cracking. Comparison of in-plane expansion to through-thickness expansion of all monitored points by plotting these data on a graph of <u>in-plane expansion</u> versus through-thickness expansion. Comparison of in-plane expansions, volumetric expansions, and through-thickness expansions recorded to date to the limits from the MPR/FSEL large-scale test programs and check of margin for future expansion.	18.2.1.31.A	At least 5 years prior to the period of extended operation and every 10 years thereafter.
67.	Structures Monitoring Program	Perform one shallow core bore in an area that was continuously wetted from borated water to be examined for concrete degradation and also expose rebar to detect any degradation such as loss of material. The removed core will also be subjected to petrographic examination for concrete degradation due to ASR per ASTM Standard Practice C856.	18.2.1.31	Complete
68.	Structures Monitoring Program	Perform sampling at the leak off collection points for chlorides, sulfates, pH and iron once every three months.	18.2.1.31	Complete

No.	PROGRAM or TOPIC	COMMITMENT	UFSAR LOCATION	SCHEDULE
69.	Open-Cycle Cooling Water System	Replace the Diesel Generator Heat Exchanger Plastisol PVC lined Service Water piping with piping fabricated from AL6XN material.	18.2.1.11	Complete
70.	Closed-Cycle Cooling Water System	Inspect the piping downstream of CC-V-444 and CC-V-446 to determine whether the loss of material due to cavitation induced erosion has been eliminated or whether this remains an issue in the primary component cooling water system.	18.2.1.12	Within ten years prior to the period of extended operation.
71.	Alkali-Silica Reaction (ASR) Monitoring Program / Building Deformation Monitoring Program	NextEra has completed testing at the University of Texas Ferguson Structural Engineering Laboratory which demonstrates the parameters being monitored and acceptance criteria used are appropriate to manage the effects of ASR. NextEra Implement the Alkali-Silica Reaction (ASR) Monitoring Program and Building Deformation Monitoring Program described in B.2.1.31A and B.2.1.31B of the License Renewal Application.	18.2.1.31A 18.2.1.31B	Prior to the period of extended operation.
72.	Flow-Accelerated Corrosion	Enhance the program to include management of wall thinning caused by mechanisms other than FAC.	18.2.1.8	Prior to the period of extended operation.
73.	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components	Enhance the program to include performance of focused examinations to provide a representative sample of 20%, or a maximum of 25, of each identified material, environment, and aging effect combinations during each 10 year period in the period of extended operation.	18.2.1.25	Prior to the period of extended operation.
74.	Fire Water System	Enhance the program to perform sprinkler inspections annually per the guidance provided in NFPA 25 (2011 Edition). Inspection will ensure that sprinklers are free of corrosion, foreign materials, paint, and physical damage and installed in the proper orientation (e.g., upright, pendant, or sidewall). Any sprinkler that is painted, corroded, damaged, loaded, or in the improper orientation, and any glass bulb sprinkler where the bulb has emptied, will be evaluated for replacement.	18.2.1.16	Prior to the period of extended operation.

No.	PROGRAM or TOPIC	COMMITMENT	UFSAR LOCATION	SCHEDULE
75.	Fire Water System	Enhance the program to a) conduct an inspection of piping and branch line conditions every 5 years by opening a flushing connection at the end of one main and by removing a sprinkler toward the end of one branch line for the purpose of inspecting for the presence of foreign organic and inorganic material per the guidance provided in NFPA 25 (2011 Edition) and b) If the presence of sufficient foreign organic or inorganic material to obstruct pipe or sprinklers is detected during pipe inspections, the material will be removed and its source is determined and corrected. In buildings having multiple wet pipe systems, every other system shall have an internal inspection of piping every 5 years as described in NFPA 25 (2011 Edition), Section 14.2.2.	18.2.1.16	Prior to the period of extended operation.
76.	Fire Water System	Enhance the Program to conduct the following activities annually per the guidance provided in NFPA 25 (2011 Edition). • main drain tests • deluge valve trip tests • fire water storage tank exterior surface inspections	18.2.1.16	Prior to the period of extended operation.
77.	Fire Water System	The Fire Water System Program will be enhanced to include the following requirements related to the main drain testing per the guidance provided in NFPA 25 (2011 Edition). • The requirement that if there is a 10 percent reduction in full flow pressure when compared to the original acceptance tests or previously performed tests, the cause of the reduction shall be identified and corrected if necessary. • Recording the time taken for the supply water pressure to return to the original static (nonflowing) pressure.	18.2.1.16	Prior to the period of extended operation.
78.	External Surfaces Monitoring	Enhance the program to include periodic inspections of in-scope insulated components for possible corrosion under insulation. A sample of outdoor component surfaces that are insulated and a sample of indoor insulated components exposed to condensation (due to the in-scope component being operated below the dew point), will be periodically inspected every 10 years during the period of extended operation.	18.2.1.24	Prior to the period of extended operation.

No.	PROGRAM or TOPIC	COMMITMENT	UFSAR LOCATION	SCHEDULE
79.	Open-Cycle Cooling Water System	Enhance the program to include visual inspection of internal coatings/linings for loss of coating integrity.	18.2.1.11	Within 10 years prior to the period of extended operation.
80.	Fire Water System	Enhance the program to include visual inspection of internal coatings/linings for loss of coating integrity.	18.2.1.16	Within 10 years prior to the period of extended operation.
81.	Fuel Oil Chemistry	Enhance the program to include visual inspection of internal coatings/linings for loss of coating integrity.	18.2.1.18	Within 10 years prior to the period of extended operation.
82.	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components	Enhance the program to include visual inspection of internal coatings/linings for loss of coating integrity.	18.2.1.25	Within 10 years prior to the period of extended operation.
83.	Alkali-Silica Reaction Monitoring	Enhance the ASR AMP to install extensometers in all Tier 3 areas of two dimensional reinforced structures to monitor expansion due to alkali-silica reaction in the out-of-plane direction. Monitoring expansion in the out-of-plane direction will commence upon installation of the extensometers and continue on a six month frequency through the period of extended operation.	18.2.1.31A	Complete
84.	ASME Section XI, Subsection IWL	Evaluate the acceptability of inaccessible areas for structures within the scope of ASME Section XI, Subsection IWL Program.	18.2.1.28	Prior to the period of extended operation.
85.	Fire Water System	Enhance the program to perform additional tests and inspections on the Fire Water Storage Tanks as specified in Section 9.2.7 of NFPA 25 (2011 Edition) in the event that it is required by Section 9.2.6.4, which states "Steel tanks exhibiting signs of interior pitting, corrosion, or failure of coating shall be tested in accordance with 9.2.7."	18.2.1.16	Prior to the period of extended operation.
86.	Fire Water System	Enhance the program to include disassembly, inspection, and cleaning of the mainline strainers every 5 years.	18.2.1.16	Prior to the period of extended operation.

No.	PROGRAM or TOPIC	COMMITMENT	UFSAR LOCATION	SCHEDULE
87.	Fire Water System	Increase the frequency of the Open Head Spray Nozzle Air Flow Test from every 3 years to every refueling outage to be consistent with LR-ISG-2012-02, AMP XI.M27, Table 4a.	18.2.1.16	Prior to the period of extended operation.
88.	Fire Water System	Enhance the program to include verification that a) the drain holes associated with the transformer deluge system are draining to ensure complete drainage of the system after each test, b) the deluge system drains and associated piping are configured to completely drain the piping, and c) normally-dry piping that could have been wetted by inadvertent system actuations or those that occur after a fire are restored to a dry state as part of the suppression system restoration.	18.2.1.16	Within five years prior to the period of extended operation.
89.	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components	Incorporate Coating Service Level III requirements into the RCP Motor Refurbishment Specification for the internal painting of the motor upper bearing coolers and motor air coolers. All four RCP motors will be refurbished and replaced using the Coating Service Level III requirements prior to entering the period of extended operation.	18.2.1.25	Prior to the period of extended operation.
90.	PWR Vessel Internals	Implement the PWR Vessel Internals Program. The program will be implemented in accordance with MRP-227-A (Pressurized Water Reactor Internals Inspection and Evaluation Guidelines) and NEI 03-08 (Guideline for the Management of Materials Issues).	18.2.1.7	Complete

No.	PROGRAM or TOPIC	COMMITMENT	UFSAR LOCATION	SCHEDULE
91	Building Deformation Monitoring	Implement the Building Deformation Monitoring Program Enhance the Structures Monitoring Program to require structural evaluations be performed on buildings and components affected by deformation as necessary to ensure that the structural function is maintained. Evaluations of structures will validate structural performance against the design basis, and may use results from the large-scale test programs, as appropriate. Evaluations for structural deformation will also consider the impact to functionality of affected systems and components (e.g., conduit expansion joints). NextEra will evaluate the specific circumstances against the design basis of the affected system or component. Enhance the Building Deformation AMP to include additional parameters to be monitored based on the results of the CEB Root Cause, Structural Evaluation and walk downs. Additional parameters monitored will include: alignment of ducting, conduit, and piping; seal integrity; laser target measurements; key seismic gap measurements; and additional instrumentation. Develop a design standard to implement Aging Management Program B.2.1.31B Building Deformation, Program Element 3 - Parameters Monitored/Inspected. The design standard will clarify the deformation evaluation process and provide an auditable format to assess it. The design standard will include steps for each of the three evaluation stages that include parameters monitored, basis for why the parameter is monitored, and conditions that prompts action for the subsequent step.	18.2.1.31B	Complete