APPENDIX A

LICENSE RENEWAL FINAL SAFETY ANALYSIS REPORT SUPPLEMENT

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A.1 Introduction

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

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

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

A.1.1 NUREG-1801 Chapter XI Aging Management Programs

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

Commitments for program additions and enhancements are identified in the Appendix A.5 License Renewal Commitment List.

1. ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (Section A.2.1.1)
2. Water Chemistry (Section A.2.1.2)
3. Reactor Head Closure Studs (Section A.2.1.3)
4. BWR Vessel ID Attachment Welds (Section A.2.1.4)
5. BWR Feedwater Nozzle (Section A.2.1.5)
6. BWR Control Rod Drive Return Line Nozzle (Section A.2.1.6)
7. BWR Stress Corrosion Cracking (Section A.2.1.7)
8. BWR Penetrations (Section A.2.1.8)
9. BWR Vessel Internals (Section A.2.1.9)
10. Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) (Section A.2.1.10)
11. Flow-Accelerated Corrosion (Section A.2.1.11)
12. Bolting Integrity (Section A.2.1.12)
13. Open-Cycle Cooling Water System (Section A.2.1.13)
14. Closed-Cycle Cooling Water System (Section A.2.1.14)
15. Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems (Section A.2.1.15)
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17. Fire Protection (Section A.2.1.17)
18. Fire Water System (Section A.2.1.18)
19. Aboveground Steel Tanks (Section A.2.1.19)
20. Fuel Oil Chemistry (Section A.2.1.20)
21. Reactor Vessel Surveillance (Section A.2.1.21)
22. One-Time Inspection (Section A.2.1.22)
23. Selective Leaching of Materials (Section A.2.1.23)
24. Buried Piping Inspection (Section A.2.1.24)
25. External Surfaces Monitoring (Section A.2.1.25)

26. Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (Section A.2.1.26)

27. Lubricating Oil Analysis (Section A.2.1.27)

28. ASME Section XI, Subsection IWE (Section A.2.1.28)

29. ASME Section XI, Subsection IWF (Section A.2.1.29)

30. 10 CFR Part 50, Appendix J (Section A.2.1.30)

31. Masonry Wall Program (Section A.2.1.31)

32. Structures Monitoring Program (Section A.2.1.32)

33. RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants (Section A.2.1.33)

34. Protective Coating Monitoring and Maintenance Program (Section A.2.1.34)

35. Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (Section A.2.1.35)

36. Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits (Section A.2.1.36)

37. Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (Section A.2.1.37)

38. Metal Enclosed Bus (Section A.2.1.38)

39. Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (Section A.2.1.39)

A.1.2 Plant-Specific Aging Management Programs

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

1. High Voltage Insulators (Section A.2.2.1)

2. Periodic Inspection (Section A.2.2.2)

3. Aboveground Non-Steel Tanks (Section A.2.2.3)

4. Buried Non-Steel Piping Inspection (Section A.2.2.4)

5. Boral Monitoring Program (Section A.2.2.5)

6. Small-Bore Class 1 Piping Inspection (Section A.2.2.6)

A.1.3 NUREG-1801 Chapter X Aging Management Programs

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

1. Metal Fatigue of Reactor Coolant Pressure Boundary (Section A.3.1.1)

2. Environmental Qualification (EQ) of Electric Components (Section A.3.1.2)

A.1.4 Time-Limited Aging Analyses

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

1. Neutron Embrittlement of the Reactor Vessel and Internals (Section A.4.2)

2. Metal Fatigue of the Reactor Pressure Vessel, Internals, and Reactor Coolant Pressure Boundary Piping and Components (Section A.4.3)

3. Environmental Qualification of Electrical Equipment (EQ) (Section A.4.4)

4. Loss of Prestress in Concrete Containment Tendons (Section A.4.5)

5. Containment Liner Plate, Metal Containment, and Penetrations Fatigue Analyses (Section A.4.6)

6. Other Plant-Specific Time Limited Aging Analyses (Section A.4.7)

A.1.5 Quality Assurance Program and Administrative Controls

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

A.2 Aging Management Programs

A.2.1 NUREG-1801 Chapter XI Aging Management Programs

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

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

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

A.2.1.2 Water Chemistry

The Water Chemistry aging management program is an existing program whose activities consist of monitoring and control of water chemistry to manage the aging of reactor vessel, reactor internals, piping, piping elements and piping components, heat exchangers and tanks that are exposed to treated water. The Water Chemistry aging management program keeps peak levels of various contaminants below system-specific limits based on industry-recognized guidelines of EPRI, BWR Vessel and Internals Project BWR Water Chemistry Guidelines for the prevention or mitigation of loss of material, reduction of heat transfer, reduction of neutron-absorbing capacity and cracking aging effects. In addition, the water chemistry program is also credited for mitigating loss of material and cracking for components exposed to sodium pentaborate, steam and reactor coolant environments. To mitigate aging effects on component surfaces the chemistry program is used to control water chemistry for impurities that accelerate corrosion.

A.2.1.3 Reactor Head Closure Studs

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

A.2.1.4 BWR Vessel ID Attachment Welds

The BWR Vessel ID Attachment Welds program is an existing aging management program that incorporates the inspection and evaluation recommendations of BWRVIP-48-A, as well as the water chemistry recommendations of BWRVIP-130. The program is implemented through station procedures that provide for mitigation of cracking through water chemistry and monitoring for cracking through in-vessel examinations of the reactor vessel internal attachment welds. Reactor vessel attachment weld inspections are implemented through station procedures that are part of in-service inspection and incorporate the requirements of ASME, Section XI.

A.2.1.5 BWR Feedwater Nozzle

The BWR Feedwater Nozzle aging management program is an existing program that manages the effects of cracking in the feedwater nozzles by enhanced inservice inspection (ISI) in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Code, Section XI, Subsection IWB, Table IWB-2500-1 and the recommendations GE-NE-523-A71-0594-A, Rev. 1. Inspections of the feedwater nozzles are performed in accordance with the Hope Creek ISI Program Plan.

A.2.1.6 BWR Control Rod Drive Return Line Nozzle

The BWR Control Rod Drive Return Line Nozzle aging management program is an existing program that provides for condition monitoring of the N9 nozzle (originally intended to be used as the CRD Return Line Nozzle) for cracking through station ISI procedures based on the ASME Section XI requirements. Hope Creek has capped the N9 nozzle to mitigate fatigue cracking. The program performs Inservice Inspection (ISI) examinations to monitor the effects of cracking on the intended function of the N9 nozzle. ISI examinations include ultrasonic inspection of the nozzle inside radius section and nozzle-to-vessel weld. Future inspections of the inside radius of the N9 nozzle will be performed using EVT-1 in accordance with NRC accepted Code case N648-1, subject to the conditions specified in Regulatory Guide 1.147. Hope Creek also conducts UT examinations of the N9 nozzle-to-cap weld in accordance with the guidelines of staff-approved boiling water reactor vessel and internals project (BWRVIP) document BWRVIP-75-A as part of the BWR Stress Corrosion Cracking program. The inspection methods used in the program have been proven effective in detecting cracking in RPV nozzles. The frequency of the inspection program is adequate to prevent significant degradation before loss of the intended function of the N9 nozzle.

A.2.1.7 BWR Stress Corrosion Cracking

The BWR Stress Corrosion Cracking aging management program is an existing program that manages intergranular stress corrosion cracking (IGSCC) in coolant pressure boundary piping and piping components made of stainless steel (SS) and nickel based alloy components as delineated in NUREG-0313, Rev. 2, and Generic Letter (GL) 88-01 and its Supplement 1. The program includes preventive measures to mitigate IGSCC, and inspection and flaw evaluation to monitor IGSCC and its effects. The schedule and extent of the inspections are performed in accordance with the NRC staff-approved BWRVIP-75-A report. To reduce the effects of IGSCC Hope Creek has also applied mechanical stress improvement process (MSIP) to several reactor pressure vessel nozzle welds. Similarly, Hope Creek has implemented Hydrogen Water Chemistry (HWC) and Noble Metals Chemical Addition (NMCA) to mitigate the environment that promotes IGSCC.

The program will be enhanced to clarify that:

1. For the components within the scope of the BWR Stress Corrosion Cracking program, resistant materials will be used for new and replacement components. This includes low carbon stainless piping and stainless steel weld material limited to a maximum carbon content 0.035 wt. % and a minimum ferrite content of 7.5%.

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

A.2.1.8 BWR Penetrations

The BWR Penetrations aging management program is an existing program that manages the effects of cracking of reactor vessel instrumentation penetrations (nozzles) exposed to reactor coolant through water chemistry and inservice inspections. The scope of the program includes beltline instrumentation nozzles and other instrumentation nozzles; except for the Standby Liquid Control/core plate differential pressure (dP) nozzle and the jet pumps instrumentation nozzles, which are in the scope of program B.2.1.1, ASME Section XI Inservice Inspection, Subsection IWB, IWC, and IWD. The BWR Penetrations aging management program incorporates the inspection and evaluation recommendations BWRVIP-49-A, “Instrument Penetration Inspection and Flaw Evaluation Guidelines,“ as well as the water chemistry recommendations of BWRVIP-130, “BWR Vessel and Internals Project BWR Water Chemistry Guidelines.”

BWRVIP-27-A addresses the Standby Liquid Control (SLC) system nozzle or housing. The guidelines of BWRVIP-27-A are applicable to plants in which the SLC system injects sodium pentaborate into the bottom head region of the vessel. The SLC system at Hope Creek injects pentaborate into the vessel through the core spray system, not through a bottom head penetration. As stated in the BWRVIP document the guidelines of BWRVIP-27-A do not apply to plants such as Hope Creek where SLC injects through the Core Spray piping. Plants that inject SLC through a bottom head penetration also used this penetration for the core plate differential pressure (dP) instrumentation. At Hope Creek inspections of the core plate dP penetration is implemented through station ISI procedures, which incorporate the requirements of ASME Section XI. The requirements of ASME Section XI are implemented in accordance with 10 CFR 50.55(a).

A.2.1.9 BWR Vessel Internals

The Hope Creek BWR Vessel Internals program is an existing program that manages the effects of cracking and loss of material of reactor pressure vessel internals through condition monitoring activities that consist of examinations implemented through station procedures consistent with the recommendations of the BWRVIP guidelines, as well as the requirements of ASME Section XI. The program also mitigates these aging effects through water chemistry activities that are implemented through station procedures that implement the guidelines of BWRVIP-130: "BWR Vessel and Internals Project BWR Water Chemistry Guidelines."

Hope Creek is committed to following the BWRVIP guidelines for managing reactor internal components. Inspections and evaluations of the Hope Creek reactor internal components are consistent with the current BWRVIP guidelines, which include but are not limited to the following BWRVIP reports:

- BWRVIP-18-A, BWR Core Spray Internals Inspection and Evaluation Flaw Guidelines.
- BWRVIP-25, BWR Core Plate Inspection and Flaw Evaluation Guidelines.
- BWRVIP-26-A, BWR Top Guide Inspection and Flaw Evaluation Guidelines.
- BWRVIP-38, BWR Shroud Support Inspection and Flaw Evaluation guidelines.
- BWRVIP 41, Jet Pump Assembly Inspection and flaw Evaluation Guidelines.
- BWRVIP-42-A LPCI Coupling Inspection and flaw Evaluation Guidelines.
- BWRVIP-47-A, BWR Lower Plenum Inspection and Flaw Evaluation Guidelines.
- BWRVIP-76, BWR Core Shroud Inspection and Flaw Evaluation Guidelines.
- BWRVIP-139, BWR Vessel and Internals Project Steam Dryer Inspection and Flaw Evaluation Guidelines.
- BWRVIP-180, BWR Vessel and Internals Project, Access Hole Cover Inspection and Flaw Evaluation Guidelines.
- BWRVIP-183, BWR Top guide Grid Beam Inspection and Flaw Evaluation Guidelines

A.2.1.10 Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)

The Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless steel (CASS) aging management program is a new program that will provide for aging management of CASS reactor internal components within the scope of license renewal. The program will include a component specific evaluation of the loss of fracture toughness in accordance with the specified criteria. For those components where loss of fracture toughness may affect function of the component, a supplemental inspection will be performed. This program will verify the integrity of the CASS components exposed to the high temperature and neutron fluence present in the reactor environment.

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

A.2.1.11 Flow-Accelerated Corrosion

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

A.2.1.12 Bolting Integrity

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

The Bolting Integrity aging management program will be enhanced to include:

1. In the following cases, bolting material should not be reused:
    a. Galvanized bolts and nuts,

b. ASTM A490 bolts; and

c. Any bolt and nut tightened by the turn of nut method.

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

A.2.1.13 Open-Cycle Cooling Water System

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

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

A.2.1.14 Closed-Cycle Cooling Water System

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

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

1. New recurring tasks will be established for enhancing the performance monitoring of the Closed Cycle Cooling Water System.

2. New recurring tasks will be established for enhancing the performance monitoring of the Chilled Water System.

3. A one-time inspection of selected Closed-Cycle Cooling Water program components in stagnant flow areas will be conducted to confirm the effectiveness of the Closed-Cycle Cooling Water program. These inspections will be performed prior to the period of extended operation.

4. A one-time inspection of selected Closed-Cycle Cooling Water program chemical mixing tanks and associated piping will be conducted to confirm the effectiveness of the Closed-Cycle Cooling Water program on the interior surfaces of the tanks and associated piping. These inspections will be performed prior to the period of extended operation.

5. The program will be enhanced such that the plant auxiliary building chilled water system, which is part of the Control Area Chilled Water System, will comply with the pure water control program in accordance with EPRI 1007820 prior to the period of extended operation.

6. A one-time inspection of selected Control Area Chilled Water System components, including the plant auxiliary building chilled water system, will be conducted to confirm the effectiveness of the Closed-Cycle Cooling Water program. These inspections will be performed prior to the period of extended operation.

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

A.2.1.15 Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems

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

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

1. The program will be enhanced to include visual inspection of structural components and structural bolts for loss of material due to general, pitting, and crevice corrosion and structural bolting for loss of preload due to self-loosening.

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

3. The acceptance criteria will be enhanced to require evaluation of significant loss of material due to corrosion for structural components and structural bolts, and significant loss of material due to wear of rail in the rail system.

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

A.2.1.16 Compressed Air Monitoring

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

A.2.1.17 Fire Protection

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

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

1. The Hope Creek routine inspection procedures will be enhanced to provide additional inspection guidance to identify degradation of fire barrier walls, ceilings, and floors for aging effects such as cracking, spalling and loss of material caused by freeze-thaw, chemical attack, and reaction with aggregates.

2. The Hope Creek fire pump supply line functional tests will be enhanced to provide specific guidance for examining exposed external surfaces of the fire pump diesel fuel oil supply line for corrosion during pump tests.

3. The Halon and Carbon Dioxide fire suppression system functional test procedures will be enhanced to include visual inspection of system piping and component external surfaces for signs of corrosion or other age related degradation, and for mechanical damage. The functional test procedures will also be enhanced to include acceptance criteria stating that identified corrosion or mechanical damage will be evaluated, with corrective action taken as appropriate.

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

A.2.1.18 Fire Water System

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

The Fire Water System program will be enhanced as follows:

1. The Fire Water System aging management program will be enhanced to inspect selected portions of the water based fire protection system piping

located aboveground and exposed to the fire water internal environment by non-intrusive volumetric examinations. These inspections shall be performed prior to the period of extended operation and will be performed every 10 years thereafter.

2. The Fire Water System aging management program will be enhanced to replace or perform 50-year sprinkler head inspections and testing using the guidance of NFPA-25 "Standard for the Inspection, Testing and Maintenance of Water-Based Fire Protection Systems" (2002 Edition), Section 5-3.1.1. These inspections will be performed prior to the 50-year in-service date and every 10-years thereafter.

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

A.2.1.19 Aboveground Steel Tanks

The Aboveground Steel Tanks aging management program is an existing program that will manage loss of material aging effects of outdoor carbon steel tanks (Fire Water Storage Tanks, Fire Diesel Fuel Oil Tank and 17-Ton CO_2 Storage Tank). Paint is a corrosion preventive measure, and periodic visual inspections will monitor degradation of the paint and any resulting metal degradation of carbon steel tanks.

The Aboveground Steel Tanks program will be enhanced as follows:

1. The program will be enhanced to include internal UT measurements to measure the wall thickness on the bottom of the tanks supported on a fiber pad on top of the concrete foundation (Fire Water Storage Tanks). Measured wall thickness will be monitored and trended if significant material loss is detected. These thickness measurements of the tank bottom will be taken and evaluated against design thickness and corrosion allowance to ensure that significant degradation is not occurring and the component intended function will be maintained during the extended period of operation.

2. The program will be enhanced to provide routine visual inspections of the carbon steel tanks external surfaces (Fire Water Storage Tanks, Fire Diesel Fuel Oil Tank and17-Ton CO_2 Storage Tank), including removal of tank insulation from the Fire Water Storage Tank to detect degradation. These inspections will be performed to detect degraded paint and coatings, and any resulting metal degradation, prior to loss of the tank intended function.

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

A.2.1.20 Fuel Oil Chemistry

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

The Fuel Oil Chemistry aging management program will be enhanced to include:

1. Equivalent requirements for fuel oil purity and fuel oil testing as described by the Standard Technical Specifications.

2. Addition of biocides, stabilizers and inhibitors as determined by fuel oil sampling or inspection activities.

3. Internal inspection of the Diesel Fire Pump Fuel Oil 280-gallon tank (T-565) using visual inspections and ultrasonic thickness examination of tank bottom.

4. Quarterly water and sediment multilevel sampling on the Diesel Fuel Oil Storage 26,500-gallon Tanks (1A-T-403, 1B-T-403, 1C-T-403, 1D-T-403, 1E-T-403, 1F-T-403, 1G-T-403, 1H-T-403) in accordance with ASTM Standard D 2709.

5. Internal inspection of the Diesel Fuel Oil Storage 26,500-gallon Tanks (1A-T-403, 1B-T-403, 1C-T-403, 1D-T-403, 1E-T-403, 1F-T-403, 1G-T-403, 1H-T-403) using visual inspections and ultrasonic thickness examination of tank bottoms.

6. Quarterly particulate sampling of the Diesel Fire Pump Fuel Oil tank (T-565) in accordance with modified ASTM 2276-00, Method A.

7. To confirm the absence of any significant aging effects, a one-time inspection of each of the 550-gallon Diesel Fuel Oil Day Tanks will be performed.

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

A.2.1.21 Reactor Vessel Surveillance

The Reactor Vessel Surveillance Program is an existing program that manages the loss of fracture toughness due to neutron irradiation embrittlement of the reactor vessel beltline materials. The program meets the requirements of 10 CFR 50, Appendix H. The program evaluates neutron embrittlement by projecting Upper Shelf Energy (USE) for reactor materials and impact on Adjusted Reference Temperature for the development of pressure-temperature limit curves. Embrittlement evaluations are performed in accordance with Regulatory Guide 1.99, Rev.2. The Hope Creek Reactor Vessel Surveillance Program is also part of the BWRVIP Integrated Surveillance Program (ISP) described in BWRVIP-86 Revision 1-A, and approved by the NRC staff. The schedule for removing surveillance capsules is in accordance with the timetable specified in BWRVIP-86 Revision 1-A for the current operating term and for the period of extended operation.

Hope Creek is a host plant for the ISP and will remove one capsule in the current operating term as part of the ISP schedule. Hope Creek will also participate in the ISP during the period of extended operation [designated as the ISP(E)]. As described in BWRVIP-86 Revision 1-A, Hope Creek's third and last capsule will be removed during the period of extended operation for testing as part of the ISP(E). If the capsule is withdrawn for some other reason, it will be stored in a manner that maintains it in a condition that would permit its future use.

The program monitors plant operating conditions to ensure appropriate steps are taken if reactor vessel exposure conditions are altered; such as the review and updating of 60-year fluence projections to support upper shelf energy calculations and pressure-temperature limit curves. The program also includes condition monitoring by removal and analysis of surveillance capsules as part of the BWRVIP ISP. These measures are effective in detecting the extent of embrittlement to prevent significant degradation of the reactor pressure vessel during the period of extended operation.

The Reactor Vessel Surveillance Program will be enhanced to include:

1. Hope Creek will implement the requirements of BWRVIP-86 Revision 1-A: "BWR Vessel and Internals Project Updated BWR Integrated Surveillance Program (ISP) Implementation Plan", October, 2012.

2. If future plant operations exceed the limitations specified in RG 1.99, the impact of plant operation changes on the extent of reactor vessel embrittlement will be evaluated and the NRC will be notified. Similarly, if future plant operation exceeds the bounds established by surveillance data that are to determine Upper Shelf Energy or P-T limits, then the impact of plant operation changes on the extent of reactor vessel embrittlement will be evaluated and the NRC will be notified. Additionally, when all the surveillance capsules are removed, then operating restrictions will be established to ensure that the plant is operated within the conditions to which the surveillance capsules were exposed. If the reactor vessel exposure conditions (neutron flux, spectrum, irradiation temperature, etc.)

are altered, then the basis for the projection to 60 years is reviewed; and, if deemed appropriate, a revised fluence projection is prepared and the effects of the revised fluence analysis on neutron embrittlement calculations will be evaluated. If necessary, an active surveillance program will be re-instituted for Hope Creek. The employment of additional surveillance specimens will be coordinated through the BWRVIP Integrated Surveillance Program (ISP). Any changes to the reactor vessel exposure conditions and the potential need to re-institute a vessel surveillance program will be discussed with the NRC staff prior to changing the plant's licensing basis.

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

A.2.1.22 One-Time Inspection

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

The One-Time Inspection aging management program will be used for the following:

1. To confirm the effectiveness of the Water Chemistry program to manage the loss of material, cracking, and the reduction of heat transfer aging effects for aluminum, copper alloy, ductile cast iron, gray cast iron, nickel alloy, steel, stainless steel and cast austenitic stainless steel in treated water, steam, sodium pentaborate, and reactor coolant environments.

2. To confirm the effectiveness of the Fuel Oil Chemistry program to manage the loss of material aging effect for copper alloy, steel, galvanized steel and stainless steel in a fuel oil environment.

3. To confirm the effectiveness of the Lubricating Oil Analysis program to manage the loss of material and the reduction of heat transfer aging effects for copper alloy, gray cast iron, steel and stainless steel in a lubricating oil environment.

4. To confirm loss of material in carbon steel piping and fittings is insignificant in an air/gas-wetted (internal) environment.

The sample plan for inspections associated with the One-Time Inspection program will be developed to ensure there are adequate inspections to address each of the material, environment, and aging effect combinations. A sample size of 20% of the population (up to a maximum of 25 inspections) will be established for each of the sample groups. Inspection methods will include visual examination or volumetric examinations. Acceptance criteria are in accordance with industry guidelines, codes, and standards, including the applicable edition of ASME Boiler and Pressure Vessel Code, Section XI. The One-Time Inspection program provides for the evaluation of the need for follow-up examinations to monitor the progression of aging if age-related degradation is found that could jeopardize an intended function before the end of the period of extended operation. Should aging effects be detected, the program triggers actions to characterize the nature and extent of the aging effect and determines what subsequent monitoring is needed to ensure intended functions are maintained during the period of extended operation.

The new program, including performance of physical inspections and evaluation of results, will be implemented prior to the period of extended operation to manage the effects of aging for selected components within the scope of license renewal.

A.2.1.23 Selective Leaching of Materials

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

One-time inspections of susceptible material and environment combinations, where selective leaching has not previously been confirmed, will be performed in the last 10 years of the current term, prior to entering the period of extended operation. A sample size of 20% of susceptible components will be subjected to a one-time inspection with a maximum of 25 inspections for each of the susceptible material groups. For material and environment combinations where selective leaching is identified, aging management activities, such as periodic inspections, will be implemented to manage aging such that the component intended function is maintained consistent with the current licensing basis through the period of extended operation.

A.2.1.24 Buried Piping Inspection

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

External inspections of buried components will occur opportunistically when they are excavated during maintenance. The Buried Piping Inspection aging management program will be enhanced to include:

1. At least one (1) opportunistic or focused excavation and inspection will be performed on each of the material groupings, which include carbon steel, ductile cast iron, and gray cast iron piping and components during each ten (10) year period, beginning ten (10) years prior to entry into the period of extended operation. A second opportunistic or focused excavation and inspection on a carbon steel piping segment, which is not cathodically protected, will be performed on the Service Water System during each ten year period, beginning ten years prior to entry into the period of extended operation. A different segment will be inspected in each ten year period.

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

A.2.1.25 External Surfaces Monitoring

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

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

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

The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components aging management program is a new program that manages the aging of the internal surfaces of steel piping, piping components and piping elements, tanks and ducting components. This program will manage the aging effect of loss of material. The program includes provisions for visual inspections of the internal surfaces of components not managed under other aging management programs. Identified deficiencies due to age related degradation are evaluated under the Corrective Action Program.

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

A.2.1.27 Lubricating Oil Analysis

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

A.2.1.28 ASME Section XI, Subsection IWE

The ASME Section XI, Subsection IWE aging management program is an existing program based on ASME Code and complies with the provisions of 10 CFR 50.55a. The program consists of periodic inspection of the primary containment surfaces and components, including its integral attachments, penetration sleeves, pressure retaining bolting, personnel airlock and equipment hatches, and other pressure retaining components for loss of material, loss of preload, and fretting or lockup.

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

The program will be enhanced to include:

1. Install an internal moisture barrier at the junction of the drywell concrete floor and the steel drywell shell prior to the period of extended operation.

2. Revise the Hope Creek ASME Section XI, Subsection IWE implementing documents to require inspection of the moisture barrier for loss of sealing in accordance with IWE 2500, after it is installed. The original design for Hope Creek did not require an internal moisture barrier at the junction of the drywell concrete floor and steel drywell shell.

3. Verify that the reactor cavity seal rupture drain lines are clear from blockage and that the monitoring instrumentation is functioning properly once prior to the period of extended operation, and one additional time during the first 10 years of the period of extended operation.

4. Establish drainage capability from the bottom of the drywell air gap on or before June 30, 2015. The drywell air gap will be divided into four approximately equal quadrants. Drainage consists of one drain in each quadrant for a total of four drains. Each drain will be open at the bottom of the drywell air gap and be capable of draining water from the air gap.

   Verify that drains at the bottom of the drywell air gap are clear from blockage once prior to the period of extended operation, and one additional time during the first 10 years of the period of extended operation.

5. Investigate the source of any leakage detected by the reactor cavity seal rupture drain line instrumentation and assess its impact on the drywell shell.

6. After drainage has been established from the bottom of the air gap from all four drains, monitor the drains at the bottom of the drywell air gap daily for leakage in the event leakage is detected by the reactor cavity seal rupture drain line instrumentation.

7. Monitor penetration sleeve J13 daily for water leakage when the reactor cavity is flooded up. In addition, perform a walkdown of the torus room to detect any leakage from other drywell penetrations. These actions shall continue until corrective actions are taken to prevent leakage through J13 or through the four air gap drains.

8. Until drainage is established from all four drains, when the reactor cavity is flooded up, perform boroscope examination of the bottom of the drywell air gap through penetrations located at elevation 93' in four quadrants, 90 degrees apart. The personnel performing the boroscope examination shall be certified as VT-1 inspectors in accordance ASME Section XI, Subsection IWA-2300, requirements. The examiners will look for signs of water accumulation and drywell shell corrosion. Adverse conditions will be documented and addressed in the corrective action program.

   After drainage has been established from the bottom of the air gap from all four drains, monitor the lower drywell air gap drains daily for water leakage when the reactor cavity is flooded up.

9. Until drainage is established from all four drains, perform UT thickness measurements each refuel outage from inside the drywell in the area of the drywell shell below the J13 penetration sleeve area to determine if there is a significant corrosion rate occurring in this area due to periodic exposure to reactor cavity leakage. In addition, UT measurements shall be performed each refuel outage around the full 360 degree circumference of the drywell between elevations 86'-11" and 88'-0" (underside of the torus down comer vent piping penetrations). Inspection and acceptance criteria

will be in accordance with IWE-2000 and IWE-3000 respectively. The results of the UT measurements shall be used to establish a corrosion rate and demonstrate that the effects of aging will be adequately managed such that the drywell can perform its intended function until April 11, 2046. Evidence of drywell shell degradation will be documented and addressed in the corrective action program.

After drainage has been established from the bottom of the air gap from all four drains, UT thickness measurements will be taken each of the next three refueling outages at the same locations as those previously examined as described above. These UT thickness measurements will be compared to the results of the previous UT inspections and, if corrosion is ongoing, a corrosion rate will be determined for the drywell shell. In the event a significant corrosion rate is detected, the condition will be entered in the corrective action process for evaluation and extent of condition determination.

10. The cause of the reactor cavity water leakage will be investigated and repaired, if practical, before PEO. If repairs cannot be made prior to the PEO, the program will be enhanced to incorporate the following aging management activities, as recommended in the Final Interim Staff Guidance LR-ISG-2006-01.

    a) Identify drywell surfaces requiring examination and implement augmented inspections for the period of extended operation in accordance with IWE-1240, as identified in Table IWE-2500-1, Examination Category E-C.

    b) Demonstrate through the use of augmented inspections that corrosion is not occurring or that corrosion is progressing so slowly that the age-related degradation will not jeopardize the intended function of the drywell shell through the period of extended operation.

    c) Develop a corrosion rate that can be inferred from past UT examinations. If degradation has occurred, evaluate the drywell shell using the developed corrosion rate to demonstrate that the drywell shell will have sufficient wall thickness to perform its intended function through the period of extended operation.

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

A.2.1.29 ASME Section XI, Subsection IWF

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

A.2.1.30 10 CFR Part 50, Appendix J

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

A.2.1.31 Masonry Wall Program

The Masonry Wall aging management program is an existing program implemented as part of the Structures Monitoring Program. The Masonry Wall Program was developed to meet the regulatory requirements of 10 CFR 50.65, Maintenance Rule, USNRC Regulatory Guide 1.160, and NUMARC 93-01, Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants. Hope Creek has no safety-related masonry walls or masonry walls whose failure during a seismic event could adversely impact a safety-related function. As a result, NRC IE Bulletin 80-11, “Masonry Wall Design”, does not apply to Hope Creek. The Masonry Wall aging management program addresses loss of material, and cracking due to age-related degradation of concrete for masonry walls. The program relies on periodic visual inspections to monitor and maintain the condition of masonry walls within the scope of license renewal.

The Masonry Wall Program will be enhanced as follows:

1. Add buildings, and masonry walls that have been determined to be in the scope of license renewal.
    a. Auxiliary Boiler Building
    b. Fire Water Pump House
    c. Masonry Wall Fire Barriers
    d. Switchyard

e. Turbine Building

2. Add an Examination Checklist for masonry wall inspection requirements.

3. Specify an inspection frequency of not greater than 5 years for masonry walls.

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

A.2.1.32 Structures Monitoring Program

The Structures Monitoring aging management program is an existing program that was developed to implement the requirements of 10 CFR 50.65 and is based on NUMARC 93-01, “Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” Rev. 2, and Regulatory Guide 1.160, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” Rev. 2. The program includes the Masonry Wall Program and the RG 1.127, Inspection of Water-Control Structures Associated With Nuclear Power Plants aging management program.

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

The Structures Monitoring Program will be enhanced to include:

1. The scope of the program will be enhanced to include the following structures and components:
    a. Auxiliary Boiler Building
    b. Fire Water Pump House
    c. Shoreline Protection Dike, and Sheet piles (Reg. Guide 1.127)
    d. Switchyard
    e. Turbine Building
    f. Transmission towers
    g. Yard Structures (Foundations for fire water tanks, manholes, transformer foundations credited for SBO)
    h. Masonry walls, including fire barriers
    i. Building penetrations that perform flood barrier, pressure boundary, shelter and protection intended functions
    j. Miscellaneous steel (catwalks, vents, louvers, platforms, etc.)
    k. Pipe whip restraints, jet impingement and missile shields
    l. Ice barrier, trash rack (Reg. Guide 1.127)
    m. Panels, racks, cabinets, and other enclosures

n. Metal-enclosed bus

o. Components supports including, electrical cable trays, electrical conduit, tubing, HVAC ducts, instrument racks, battery racks, and supports for piping and components that are not within the scope of ASME Section XI, Subsection IWF

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

2. Concrete structures will be observed for a reduction in equipment anchor capacity due to local concrete degradation. This will be accomplished by visual inspection of concrete surfaces around anchors for cracking and spalling.

3. Clarify that inspections are performed for loss of material due to corrosion and pitting of additional steel components, such as embedments, panels and enclosures, doors, siding, metal deck, and anchors.

4. Perform a one-time inspection of the external stainless steel surfaces of the expansion bellows at Condensate Storage Tank Dike for loss of material due to corrosion, within the ten-year period prior to the period of extended operation.

5. Require inspection of penetration seals, structural seals, and elastomers, for degradations that will lead to a loss of sealing by visual inspection of the seal for hardening, shrinkage and loss of strength.

6. Require monitoring of vibration isolators, associated with component supports other than those covered by ASME XI, Subsection IWF.

7. Add an Examination Checklist for masonry wall inspection requirements.

8. Parameters monitored for wooden components will be enhanced to include: change in material properties, loss of material due to insect damage and moisture damage.

9. Specify an inspection frequency of not greater than 5 years for structures including submerged portions of the Service Water Intake Structure.

10. Require individuals responsible for inspections and assessments for structures to have a B.S. Engineering degree and/or Professional Engineer license, and a minimum of four years experience working on building structures.

11. Perform periodic sampling, testing, and analysis of ground water chemistry for pH, chlorides, and sulfates on a frequency of 5 years.

12. Require supplemental inspections of the in scope structures within 30 days following extreme environmental or natural phenomena (large floods, significant earthquakes, hurricanes, and tornadoes).

13. Perform a chemical analysis of ground or surface water in-leakage when there is significant in-leakage or there is reason to believe that the in-leakage may be damaging concrete elements or reinforcing steel.

14. Implementing procedures will be enhanced to include additional acceptance criteria details specified in ACI 349.3R-96.

These enhancements will be implemented prior to the period of extended operation. The one-time inspection in enhancement 4 will be performed within the ten-year period prior to the period of extended operation.

A.2.1.33 RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants

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

The RG 1.127, Inspection of Water-Control Structures Associated With Nuclear Power Plants Program will be enhanced to include:

1. Shoreline Protection and Dike structures will be added to the program.

2. Parameters monitored for wooden components will be enhanced to include change in material properties and loss of material due to insect damage and moisture damage.

3. The inspection requirement for submerged concrete structural components will be enhanced to require that inspections be performed by dewatering a pump bay or by a diver if the pump bay is not dewatered.

4. Specify an inspection frequency of not greater than 5 years for structures including submerged portions of the Service Water Intake Structure.

5. Require supplemental inspections of the in scope structures within 30 days following extreme environmental or natural phenomena (large floods, significant earthquakes, hurricanes, and tornadoes).

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

A.2.1.34 Protective Coating Monitoring and Maintenance Program

The Protective Coating Monitoring and Maintenance Program is an existing program that provides for aging management of Service Level 1 coatings inside the drywell and torus. Service Level 1 coatings are used in areas where coating failure could adversely affect the operation of post-accident fluid systems and thereby impair safe shutdown. The Protective Coating Monitoring and Maintenance Program provides for inspections, assessments, and repairs for any condition that adversely affects the ability of Service Level 1 coatings to function as intended.

A.2.1.35 Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

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

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

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

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

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

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

A.2.1.37 Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

The Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements aging management program is a new program that will be used to manage the aging effects and mechanisms of non-EQ, in scope inaccessible power cables (480 volts, 4,160 volts and 13,800 volts). These cables may at times be exposed to significant moisture simultaneously with significant voltage. Significant moisture is defined as periodic exposures that last more than a few days (e.g., cable in standing water). Periodic exposures that last less than a few days (e.g., normal rain and drain) are not significant. Significant voltage exposure is defined as being subject to system voltage for more than twenty-five percent of the time. Note that no inaccessible power cable exposed to significant moisture was excluded from the program due to the “significant voltage” criterion. The Hope Creek cables in the scope of this aging management program will be tested using a proven test for detecting deterioration of the insulation system due to wetting that is state-of-the-art at the time the test is performed. The cable test frequency will be established based on test results and industry operating experience. The maximum time between tests will be no longer than 6 years. The first tests will be completed prior to the period of the extended operation.

Prior to the period of extended operation, manholes and cable vaults associated with the cables included in this aging management program will be inspected for water collection (with water removal as necessary). The objective of the inspections, as a preventive action, is to minimize exposure of the power cables to significant moisture. The frequency of inspections for accumulated water will be established based on inspection results. This approach to determining inspection frequency recognizes that a recurring inspection, set at the optimum frequency, would result in the cables being submerged only as a result of event driven, rain and drain, type occurrences. Station procedures will direct the assessment of the cable condition as a result of rain or other event driven occurrences. As a limit on the amount of time between inspections, the maximum time between inspections will be no more than 1 year.

The Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements aging management program will be enhanced as follows:

1. Add low voltage power cables (480 volts or greater) to the scope of the

program.

2. Change cable testing maximum frequency from 10 years to 6 years. Change cable vault and manhole inspection maximum frequency from 2 years to 1 year.

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

A.2.1.38 Metal Enclosed Bus

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

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

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

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

A.2.1.39 Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

The Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program is a new program that will be used to confirm the absence of an aging effect with respect to electrical cable connection stressors. A representative sample of non-EQ electrical cable connections will be selected for one-time testing considering application (medium and low voltage), circuit loading (high loading) and location, with respect to connection stressors. The technical basis for the sample selected will be documented. The specific type of test performed will be a proven test for detecting loose connections, such as thermography or contact resistance measurement, as appropriate to the application.

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

A.2.2 Plant-Specific Aging Management Programs

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

A.2.2.1 High Voltage Insulators

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

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

A.2.2.2 Periodic Inspection

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

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

A.2.2.3 Aboveground Non-Steel Tanks

The Aboveground Non-Steel Tanks aging management program is a new program that will manage loss of material of outdoor non-steel tanks in the scope of license renewal. The tank within the scope of this program is the stainless steel condensate storage tank. Periodic visual inspections will monitor for degradation of the tank external surface. Periodic visual inspections will also monitor for degradation of the seal at the interface between the tank bottom and the concrete foundation.

The Aboveground Non-Steel Tanks program will include a UT wall thickness inspection of the bottom of the tank. The UT measurements will be taken to ensure that significant degradation is not occurring and that the component intended function will be maintained during the extended period of operation.

This new program will be implemented prior to the period of extended operation. Tank bottom UT inspections will also be performed prior to the period of extended operation.

A.2.2.4 Buried Non-Steel Piping Inspection

The Buried Non-Steel Piping Inspection aging management program is an existing condition monitoring program that manages the buried reinforced concrete piping and components in the Service Water System that are exposed to an external soil or groundwater environment for cracking, loss of bond, increase in porosity and permeability and loss of material. These aging effects will be identified through visual inspections of the external surfaces of the piping and components.

The Buried Non-Steel Piping Inspection aging management program also inspects the buried stainless steel piping and components in the Condensate Storage and Transfer System and the Fire Protection System for loss of material. These aging effects will be identified through visual inspections of the external surfaces of the piping and components. The Buried Non-Steel Piping Inspection will be enhanced to include:

1. At least one (1) opportunistic or focused excavation and inspection will be performed on buried reinforced concrete piping and components during each ten (10) year period, beginning ten (10) years prior to entry into the period of extended operation.

2. At least one (1) opportunistic or focused excavation and inspection will be performed on Condensate Storage and Transfer System buried stainless steel piping and components, which contain fluid that exceed EPA drinking water limits, during each ten (10) year period, beginning ten (10) years prior to entry into the period of extended operation.

3. Guidance for inspection of concrete aging effects.

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

A.2.2.5 Boral Monitoring Program

The Boral Monitoring Program is an existing program that monitors the Boral test coupon inspection and/or testing results at other boiling water reactor (BWR) sites. If these results indicate a problem with Boral neutron absorbing material potentially affecting its intended function (i.e., absorb neutrons), Hope Creek will initiate inspection and/or testing of its Boral test coupons in the Hope Creek spent fuel pool (SFP). The Boral Monitoring Program will be enhanced to include:

1. Inspection, testing and evaluation of one coupon from the Hope Creek spent fuel pool prior to the period of extended operation and one coupon within the first 10 years after entering the period of extended operation. Testing will include dimensional and neutron attenuation measurements with an acceptance criteria of no more than a 10% increase in thickness and no more than a 5% decrease in B-10 areal density.

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

A.2.2.6 Small-Bore Class 1 Piping Inspection

The Small-Bore Class 1 Piping Inspection program is a new program that will manage the aging effect of cracking in small-bore (greater than or equal to NPS 1 and less than NPS 4) Class 1 piping through the use of a combination of volumetric examinations and visual inspections. This new program is comprised of the existing ASME Section XI ISI (Risk Informed Inservice Inspection, RI-ISI) program that performs volumetric and visual examinations for selected Class 1 small-bore butt welds and other selected small-bore socket welds, and supplemental inspections consisting of 25 Class 1 small-bore socket welds and 25 Class 1 small-bore butt welds using volumetric or other industry approved techniques. The RI-ISI program provides a robust inspection selection process and is based upon the susceptibility to degradation and the consequences of a piping failure, which is founded on actual service experience with nuclear plant piping failure data. The RI-ISI program requires volumetric and VT-2 examinations on a frequency and number determined by ASME Code Case N-578-1 and the Hope Creek ISI Program Plan. These ongoing inspections combined with supplemental inspections consisting of 25 Class 1 small-bore socket welds and 25 Class 1 small-bore butt welds using volumetric or other industry approved techniques and based on a sample methodology to select the most susceptible and risk-significant welds for inspection will be effective in identifying any age related or underlying deficiencies. Any deficiencies identified are evaluated under the corrective action program.

The Small-Bore Class 1 Piping Inspection program will effectively manage the aging effect of cracking in small-bore (greater than or equal to NPS 1 and less than NPS 4) Class 1 piping by identifying and evaluating cracking prior to loss of intended function.

This new program will be implemented prior to the period of extended operation, with the supplemental inspections performed within the six year period prior to the period of extended operation.

A.3 NUREG-1801 Chapter X Aging Management Programs

A.3.1 Evaluation of Chapter X Aging Management Programs

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

A.3.1.1 Metal Fatigue of Reactor Coolant Pressure Boundary

The Metal Fatigue of Reactor Coolant Pressure Boundary Program is an existing program that manages cumulative fatigue damage in the selected reactor coolant components subject to the reactor coolant and treated water environments.

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

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

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

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

1. The Metal Fatigue of Reactor Coolant Pressure Boundary program will be enhanced to include additional transients beyond those defined in the Technical Specifications and the UFSAR, and expanding the fatigue monitoring program to encompass other components identified to have fatigue as an analyzed aging effect, which require monitoring.
2. The Metal Fatigue of Reactor Coolant Pressure Boundary program will be enhanced to use a software program to automatically count transients and calculate cumulative usage on select components. At this time only cycle based fatigue monitoring will be used. If stress based fatigue monitoring is used in the future, it will consider the six stress terms in accordance with the methodology from ASME Section III, Subsection NB, Subarticle NB-3200.
3. The Metal Fatigue of Reactor Coolant Pressure Boundary program will be enhanced to address the effects of the reactor coolant environment on component fatigue life by assessing the impact of the reactor coolant environment on a sample of critical components for the plant identified in NUREG/CR-6260.
4. The Metal Fatigue of Reactor Coolant Pressure Boundary program will be enhanced to require a review of additional reactor coolant pressure boundary locations if the usage factor for one of the environmental fatigue sample locations approaches its design limit.

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

A.3.1.2 Environmental Qualification (EQ) of Electric Components

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

A.4 Time-Limited Aging Analyses

A.4.1 Introduction

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

A.4.2 Neutron Embrittlement of the Reactor Pressure Vessel and Internals

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

High energy neutron fluence for the welds and shells of the reactor pressure vessel beltline region was calculated using the RAMA fluence methodology. The RAMA methodology was developed for the Electric Power Research Institute and the Boiling Water Reactor Vessel and Internals Project and is approved by the NRC for use at Hope Creek. Use of this methodology for evaluations of fluence for Hope Creek was performed in accordance with guidelines presented in Regulatory Guide 1.190. The evaluations determined values for neutron fluence for extended power uprate (EPU) conditions and for extended operation out to 56 effective full power years (EFPY), i.e., at the end of 60 years of operation.

A.4.2.1 Reactor Pressure Vessel Materials Upper-Shelf Energy Reduction Due to Neutron Embrittlement

The regulations governing reactor pressure vessel integrity are in 10 CFR 50. 10 CFR 50.60 requires that all light-water reactors meet the fracture toughness, pressure-temperature limits, and material surveillance program requirements for the reactor coolant pressure boundary as set forth in 10 CFR 50 Appendices G and H. The Hope Creek current licensing basis analyses evaluating reduction of fracture toughness of the reactor pressure vessel for 40 years are TLAAs. 10 CFR 50 Appendix G requires the predicted end-of-life Charpy impact test USE for reactor pressure vessel materials to be at least 50 ft-lb (absorbed energy), unless an approved analysis supports a lower value. The predicted USE drop is determined in accordance with RG 1.99. Predicted USE drop for each reactor pressure vessel material in the beltline region exposed to fluence greater than 1.0×10^{17} n/cm^2 for 56 EFPY was determined in accordance with RG 1.99, Revision 2.

The 50 ft-lb criterion was met for all materials evaluated and therefore, the analyses are projected for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A.4.2.2 Adjusted Reference Temperature for Reactor Pressure Vessel Materials Due to Neutron Embrittlement

The adjusted reference temperature (ART) of the limiting beltline material is used to adjust the beltline P-T curves to account for irradiation effects. RG 1.99 Revision 2 provides the methods for determining the ART. The initial nil-ductility reference temperature, RT_{NDT}, is the temperature at which a non-irradiated metal (ferritic steel) changes in fracture characteristics going from ductile to brittle behavior. RT_{NDT} is evaluated according to the procedures in the ASME Code, Section III. 10 CFR 50 Appendix G defines the fracture toughness requirements for the life of the vessel. The ART is defined as Initial RT_{NDT} + ΔRT_{NDT} + Margin. The Margin term is defined in RG 1.99 Revision 2. The current P-T curves are developed from limiting ART values for the vessel materials and are valid for 32 EFPY. Since the ART values are determined by adding the unirradiated RT_{NDT} and the ΔRT_{NDT} values for the 40-year licensed operating period, these ART calculations meet the criteria of 10 CFR 54.3(a) and have been identified as TLAAs requiring evaluation for 60 years.

56 EFPY fluence values were calculated for the Hope Creek reactor pressure vessel for the extended 60-year licensed operating period, using the RAMA Fluence Methodology software package. The 56 EFPY ΔRT_{NDT} values for all beltline materials (i.e., those materials with a fluence exposure greater than $1x10^{17}$ n/cm^2) were calculated based on the embrittlement correlation found in RG 1.99. Hope Creek will re-project, as necessary, the 56 EFPY ΔRT_{NDT} and ART values in conjunction with the surveillance capsule results from the BWRVIP Integrated Surveillance Program (ISP) as described in BWRVIP-86 Revision 1-A.

The analysis for the adjusted reference temperature has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A.4.2.3 Reactor Pressure Vessel Analyses: Pressure – Temperature Limits

10 CFR Part 50 Appendix G requires reactor pressure vessel analyses to determine operating P-T limits for boltup, hydrotest, pressure tests and normal operating and anticipated operational occurrences. The ART of the limiting beltline material is used to adjust the beltline P-T limits to account for irradiation effects. Operating limits for pressure and temperature are required for three categories of operation: 1) hydrostatic pressure tests and leak tests, referred to as Curve A; 2) non-nuclear heat-up / cooldown and low-level physics tests, referred to as Curve B; and 3) core critical operation, referred to as Curve C. P-T limits are developed for three bounding vessel regions: the upper vessel region (non-beltline, including the head flange region), the core beltline region, and the vessel bottom head region. The calculations associated with generation of the P-T curves satisfy the criteria of 10 CFR 54.3(a). As such, these calculations are a TLAA.

The current HCGS Technical Specifications contain P-T curves for hydrostatic pressure and leak tests, non-nuclear heat-up/cooldown and low-level physics tests, and critical operation. Technical Specifications also limit the maximum rate of change of the reactor coolant temperature. These P-T curves provide limits up to 32 EFPY. The P-T curves were developed to present steam dome pressure versus minimum vessel metal temperature, incorporating appropriate non-beltline limits and irradiation embrittlement effects in the beltline. Because of the relationship between the P-T limits and the fracture toughness transition of the HCGS reactor pressure vessel, new P-T limits are required to be calculated and will be submitted as updates to the NRC, in compliance with 10 CFR Part 50, Appendix G.

HCGS will re-project the P-T curves using approved fluence calculations when there are changes in power or significant changes in core design in conjunction with surveillance capsule results from the BWRVIP Integrated Surveillance Program (ISP) (BWRVIP-86 Revision 1-A).

Reactor Pressure Vessel Analyses for P-T limits, considering the effects of aging on the intended function will be adequately managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

A.4.2.4 Reactor Pressure Vessel Circumferential Weld Examination Relief

ASME Code, Section XI governs inspection of the reactor pressure vessel circumferential welds, as implemented by the Hope Creek In-service Inspection Program. These welds are required to be inspected at regular intervals described in Table IWB-2500-1, Examination Category B-A of ASME Code, Section XI. Hope Creek has received inspection relief for the circumferential welds for the time remaining in the current 40 year licensed operating period. This inspection relief is based upon NRC Generic Letter 98-05, which is based on probabilistic assessments that predict an acceptably low probability of failure per reactor operating year.

The analysis is based on reactor pressure vessel metallurgical conditions as well as flaw indication sizes and frequencies of occurrence that are expected through the end of the current licensed operating period. The basis for this relief request was an analysis that satisfied the limiting conditional failure probability for the circumferential welds at the expiration of the current license, based on BWRVIP-05 and the extent of neutron embrittlement expected through 40 years of operation. The anticipated metallurgical effects due to increased fluence expected during the period of extended operation require evaluation for 56 EFPY (corresponding to 60 years) and approval by the NRC to extend this relief request. The circumferential volumetric weld examination relief analysis meets the requirements of 10CFR54.3(a) and is a TLAA.

The failure frequency of circumferential welds for Hope Creek was calculated for Hope Creek for the period of extended operation and was projected to be less than the criteria established in the NRC's SER for BWRVIP-05. A request for extension of this relief for Hope Creek for the extended operating period will be submitted to the NRC, in accordance with 10 CFR 50.55(a), prior to the period of extended operation.

The analysis for the circumferential weld examination relief has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A.4.2.5 Reactor Pressure Vessel Axial Weld Failure Probability

The BWRVIP recommendations for inspection of reactor pressure vessel shell welds in BWRVIP-05 contain generic analyses supporting a conclusion in the NRC SER that the generic-plant axial weld failure rate is no more than 5×10^{-6} per reactor year.

The probability of a failure event, PoF, calculated by the NRC in the BWRVIP-05 SER and its supplements depends in part on an assumption that essentially 100% (≥90%) of axial welds can be inspected. At Hope Creek, greater than 90% of the axial welds can be examined.

The anticipated changes in metallurgical conditions expected over the extended licensed operating period require an additional analysis for 56 EFPY. As such, this analysis meets the requirements of 10 CFR 54.3(a) and has been identified as a TLAA requiring evaluation for the period of extended operation.

Since the comparison of neutron fluence, initial RT_{NDT}, chemistry factor amounts of copper and nickel, delta RT_{NDT}, and the projected mean RT_{NDT} of the limiting axial weld at the end of the period of extended operation has been projected to be less than that found in the reference case in the BWRVIP and NRC analyses, the axial weld failure probability continues to be bounded by the previously approved value of less than 5E-6 per reactor year.

The analysis for the axial weld failure probability has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A.4.2.6 Reactor Pressure Vessel Core Reflood Thermal Shock Analysis

General Electric Report No. NEDO-10029 addressed the concern for brittle fracture of the reactor pressure vessel due to reflood following a postulated Loss of Coolant Accident (LOCA). The thermal shock analysis documented in the report assumed a design basis LOCA followed by a low pressure coolant injection accounting for the full effects of neutron embrittlement at the end of 40 years. This analysis bounded only 40 years of operation. Therefore, reflood thermal shock of the reactor pressure vessel is considered a TLAA for the Hope Creek.

Since the time of the NEDO-10029 analysis, another analysis has been performed for BWR-6 vessels in “Fracture Mechanics Evaluation of a Boiling Water Vessel Following a Postulated Loss of Coolant Accident”. The more recent BWR-6 analysis was reevaluated for the Hope Creek BWR-4 reactor pressure vessel and found to be bounding for the period of extended operation. Because the available fracture toughness was found to be greater than the peak stress intensity calculated for the event evaluated, brittle fracture of the Hope Creek reactor pressure vessel due to vessel reflood following a design basis LOCA is not possible during the period of extended operation.

The core reflood thermal shock analysis relief has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A.4.2.7 Reactor Internals Components

A number of the reactor internals components are subject to high fluence because of their proximity to the core. This high fluence can lead to stress relaxation for bolting. Stress relaxation of the core plate rim hold-down bolts is a TLAA issue.

A plant specific analysis demonstrates that the pre-load of the core plate rim hold-down bolts is sufficient to prevent lateral motion of the core plate until the plant reaches 43.9 EFPY. Prior to the period of extended operation, which is prior to the analyzed limit of 43.9 EFPY, the plant will either; (1) install core plate wedges or (2) perform an analysis that demonstrates the component function is maintained.

The core plate rim hold-down bolts TLAA will be dispositioned in accordance with 10 CFR 54.21(c)(1)(iii), by managing the aging effects with the analysis that projects loss of preload at 43.9 EFPY and taking action as described above, prior to the period of extended operation.

A.4.3 Metal Fatigue of the Reactor Pressure Vessel, Internals, and Reactor Coolant Pressure Boundary Piping and Components

Metal fatigue was considered explicitly in the design process for pressure boundary components designed in accordance with ASME Section III, Class A or Class 1 requirements. Metal fatigue was evaluated implicitly for components designed in accordance with ASME Section III, Class 2 or 3 requirements or ANSI B31.1 requirements. Each of these fatigue analyses and evaluations are considered to be Time-Limited Aging Analyses (TLAAs) requiring evaluation for the period of extended operation in accordance with 10 CFR 54.21(c).

A.4.3.1 Reactor Pressure Vessel Fatigue Analyses

Reactor pressure vessel fatigue analyses depend on the assumed numbers and the severity of normal and upset event pressure and thermal operating cycles to predict end-of-life fatigue usage factors in accordance with Section III of the ASME Code. These assumed cycle counts used to determine fatigue usage factors are based on the 40-year life of the plant. The calculation of fatigue usage factors is part of the current licensing basis and is used to support safety determinations. As such the reactor pressure vessel fatigue analyses meet the requirements of 10 CFR 54.3(a) for a TLAA.

The original Hope Creek reactor pressure vessel stress report included fatigue analyses for the reactor pressure vessel components based on a set of design basis duty cycles. The original 40-year analyses demonstrated that the cumulative usage factors (CUFs) for all critical components would remain below the allowable fatigue usage value of 1.0 specified in Section III of the ASME Code.

Revised CUF evaluations were also performed to assess the impact of Extended Power Uprate (EPU) on the reactor pressure vessel for Hope Creek. These evaluations revised the CUF values for some reactor pressure vessel components based on the change in reactor operating conditions resulting from EPU. The revised CUF evaluations were approved by the NRC as a part of the EPU approval process.

In addition, revised fatigue evaluations were performed for the recirculation inlet nozzle for containment or "new" loads in 1988, for the top head and vessel flanges for asymmetric head spray cooling in 1990, for the main closure region to support reduced-pass stud tensioning in 2000, for the reactor pressure vessel support skirt to evaluate cumulative fatigue loadings in 1996, and for the core spray nozzles to evaluate the impact of additional High Pressure Coolant Injection (HPCI) injections in 2008.

The list of design transients used in the reactor pressure vessel fatigue analyses was intended to envelope all anticipated thermal and pressure cycles that could be expected to occur within a nominal 40-year operating period for the plant. The number of transients experienced to-date for the reactor pressure vessel and other analyzed components was compiled from the Hope Creek Cycle Counting Program. The numbers of occurrences expected for 40 and 60 years of operation were obtained by extrapolating the numbers of occurrences actually incurred to-date, and using the rate of occurrence experienced during the last twelve years of operation (nine operating cycles). Conservatism was added beyond the mathematically projected number of cycles to accommodate potential variation in plant performance late in plant life, as well as to allow for additional events where the projected number of cycles was very low and the likelihood of additional events could not be ruled out.

The CUFs of the reactor pressure vessel, including the vessel support skirt, shell, upper and lower heads, closure flanges, nozzles and penetrations, nozzle safe ends, refueling bellows support, and closure studs will be managed by the Metal Fatigue of Reactor Coolant Pressure Boundary aging management program. The program includes monitoring and recording transient events and periodic updating to current and projected CUF values.

All locations with CUF ratios (i.e., CUF/allowable) predicted to exceed 0.4 (or 40% of allowable) in the original design basis fatigue analysis will be included in the program. In addition, the locations identified in NUREG/CR-6260 for the newer-vintage General Electric plant, which have been evaluated for environmental fatigue effects have been included in the program.

The Metal Fatigue of Reactor Coolant Pressure Boundary program will monitor the numbers of cycles of the design transients and the corresponding CUF for critical reactor pressure vessel components. All necessary plant transient events will be tracked to ensure that the CUF remains less than the allowable CUF limit for all monitored components. In the event the monitored CUF is predicted to exceed the allowable value for any component prior to 60 years of operation, appropriate corrective action will be taken in accordance with the corrective action process prior to the allowable limits being exceeded.

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

A.4.3.2 Reactor Pressure Vessel Internals Fatigue Analyses

The Hope Creek reactor pressure vessel internals were not designed in accordance with the codes for the reactor pressure vessel. However, the Hope Creek UFSAR reports CUF values for three reactor pressure vessel internals components as follows:

(1) 0.111 for the core support plate (at stud),

(2) 0.435 for the top guide (at beam slot), and

(3) Less than 0.05 for the core differential pressure sensing line (at elbow).

As such, the reactor pressure vessel internals fatigue analyses have been identified as TLAAs that meet the requirements of 10 CFR 54.3(a).

The reactor pressure vessel internals are not a part of the Class 1 pressure boundary. The above calculated fatigue usage factors when extrapolated to 60 years indicate acceptable fatigue performance for the period of extended operation.

The analysis has been projected through the period of extended operation and found to be acceptable in accordance with 10 CFR 54.21(c)(1)(ii).

A.4.3.3 Reactor Coolant Pressure Boundary Piping and Component Fatigue Analyses

The Hope Creek reactor coolant pressure boundary (RCPB) piping was designed in accordance with ASME Code, Section III, Class 1, as stated in Section 3.2.2 of the Hope Creek UFSAR.

The Hope Creek piping systems in the scope of license renewal designed to ASME Code, Section III, Class 1 requirements were explicitly analyzed for fatigue. The analyses demonstrated that the 40-year CUFs for the limiting components in the affected systems were below the ASME Code Section III allowable value of 1.0.

High energy line breaks (HELB) in the Hope Creek piping have been postulated, and based on analyses performed for these breaks, means for avoiding damage to surrounding equipment and systems have been incorporated in the plant. Potential damage could arise from fluid jet impingement, flooding, compartment pressurization, environmental effects, and pipe whip. It is noted that cumulative fatigue usage factors (CUF <0.1) for the high energy lines are included in the criteria to determine postulated breaks. The CUFs, as calculated in the design fatigue analyses, account for the design transients assumed for the original 40-year life of the plant. The CUF calculations used in the selection of postulated high energy line break locations are TLAAs.

Since these breaks are postulated to occur only once in the lifetime of the plant, and restraints were installed appropriately to mitigate these potential breaks, the results of analyses for the potential breaks and the restraints installed in the plant remain unchanged for the extended life of 60 years. However, it is possible that other locations that had 40-year CUFs below the criteria for postulated breaks could exceed the CUF criteria in 60 years. The possibility of these additional postulated breaks will need to be managed based on the actual fatigue accumulation encountered as the plant ages.

The Hope Creek RCPB piping stress reports include fatigue analyses for all Class 1 piping components based on a set of design basis duty cycles. The original 40-year analyses demonstrated that the CUFs for all critical piping components would remain below the allowable fatigue usage value of 1.0 specified in Section III of the ASME Code.

The HELB locations were considered as a part of the Class 1 piping screening evaluation for identifying locations to monitor in the Hope Creek Metal Fatigue of Reactor Coolant Pressure Boundary program. As such, all critical HELB locations are encompassed by the Hope Creek Metal Fatigue of Reactor Coolant Pressure Boundary program. Revised CUF evaluations have been performed on the piping since the time of original plant construction. These evaluations revised the CUF values for some piping components based on the change in piping configuration or the change in reactor and piping operating conditions. The CUFs of the piping will be managed by the Metal Fatigue of Reactor Coolant Pressure Boundary aging management program. This program will monitor critical piping component cycle based fatigue (CBF) through the use of a fatigue monitoring software application, using CBF monitoring versus the allowable value. All bounding Class 1 piping locations with CUF ratios (i.e., CUF/allowable) predicted to exceed 0.4 (or 40% of allowable) in the original design basis fatigue analysis will be included in the program. In addition, the locations identified in NUREG/CR-6260 for the newer-vintage General Electric plant, which have been evaluated for environmental fatigue effects have been included in the program.

The Metal Fatigue of Reactor Coolant Pressure Boundary program will monitor the numbers of cycles of the design transients and the corresponding CUF for critical Class 1 piping components including high energy line break limiting locations. In the event the monitored CUF is predicted to exceed the allowable value for any component prior to 60 years of operation, appropriate corrective action will be taken in accordance with the corrective action process prior to the allowable limits being exceeded. As such, the Metal Fatigue of Reactor Coolant Pressure Boundary program will manage the effects of aging due to fatigue on the RCPB piping in accordance with 10 CFR 54.21(c)(1)(iii).

A.4.3.4 Non-Class 1 Component Fatigue Analyses

Non-Class 1 components include pipe, tubing, fittings, tanks, vessels, heat exchangers, valve bodies, pump casings, and miscellaneous process components. These components were generally designed in accordance with appropriate ASME Section III subsections or American National Standards Institute (ANSI) B31.1, or B31.7, depending on their function.

Calculation of cumulative fatigue usage is not required for non-Class 1 components. For non-Class 1 components, stresses due to thermal expansion and anchor movement, which are important for fatigue evaluations, are analyzed using stress intensification factors and stress allowables. Allowable stresses are defined for 7,000 full temperature cycles with reductions in allowable stresses as cycles increase beyond 7,000.

For the Hope Creek piping systems in the scope of license renewal designed to B31.1, or ASME Code, Section III, Class 2 and 3, the assumed thermal cycle count can conservatively be approximated by the thermal cycles expected for the reactor pressure vessel, which is less than 2,700 for 60 years of plant operation. This is a fraction of the 7,000-cycle threshold. Therefore, the existing piping analyses designed to B31.1, or ASME Code, Section III, Class 2 and 3 within the scope of license renewal are valid for the period of extended operation in accordance 10 CFR 54.21(c)(1)(i).

A.4.3.5 Effects of Reactor Coolant Environment on Fatigue Life of Components and Piping (Generic Safety Issue 190)

The NRC concluded that licensees who apply for license renewal should address the effects of coolant environment on component fatigue life as part of their aging management programs. 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 Section X.M1, “Metal Fatigue of Reactor Coolant Pressure Boundary,” requires detailed, vintage-specific, fatigue calculations for plants applying for license renewal for the locations identified in NUREG/CR-6260, “Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components”. Since such calculations do not form a part of the Hope Creek current licensing basis and therefore do not satisfy the requirements of 10 CFR 54.3, they are not TLAA.

From NUREG/CR-6260 for the newer-vintage General Electric plant, the following locations require evaluation:

- Reactor pressure vessel shell and lower head

- Reactor pressure vessel feedwater nozzle

- Reactor recirculation piping (including reactor pressure vessel inlet and outlet nozzles)
- Core spray line reactor pressure vessel nozzle and associated Class 1 piping
- Residual heat removal nozzles and associated Class 1 piping
- Feedwater Class 1 piping

The cumulative usage factors (CUF) were calculated for each of these locations, accounting using the appropriate environmental fatigue multiplier, or F_{en}, relationships from NUREG/CR-6583 for carbon/low alloy steels, and NUREG/CR-5704 for stainless steels, as appropriate for the material for each location. The results demonstrate that CUF values, including appropriate environmental effects, are less than 1.0 for 60 years of plant operation for all locations except for the reactor pressure vessel feedwater nozzle safe end. Corrective action will be taken prior to exceeding the environmental assisted fatigue CUF value of 1.0.

All of the locations discussed will be included in the Metal Fatigue of Reactor Coolant Pressure Boundary aging management program, and the CUF for these locations will be tracked in accordance with 10 CFR 54.21(c)(1)(iii).

A.4.4 Environmental Qualification of Electrical Equipment

Thermal, radiation, and cyclical aging analyses of plant electrical and I&C equipment, developed to meet 10 CFR 50.49 requirements, have been identified as time-limited aging analyses (TLAAs) for Hope Creek. The NRC has established nuclear station environmental qualification (EQ) requirements in 10 CFR 50.49 and 10 CFR 50, Appendix A, Criterion 4. 10 CFR 50.49 specifically requires that an EQ program be established to demonstrate that certain electrical components located in harsh plant environments are qualified to perform their safety function in those harsh environments after the effects of in-service aging. Harsh environments are defined as those areas of the plant that could be subject to the harsh environmental effects of a loss-of-coolant accident (LOCA), high energy line break (HELB), or post-LOCA radiation. 10 CFR 50.49 requires that the effects of significant aging mechanisms be addressed as part of environmental qualification. The qualification evaluations of electrical equipment with a qualified life of at least 40 years are considered TLAA for license renewal.

The Hope Creek Environmental Qualification (EQ) of Electric Components program will manage the effects of aging effects for the components associated with the environmental qualification TLAA. The reanalysis of an aging evaluation is normally performed to extend the qualification by reducing excess conservatism incorporated in the prior evaluation. Reanalysis of an aging evaluation to extend the qualification of a component is performed on a routine basis pursuant to 10 CFR 50.49(e) as part of the Hope Creek Environmental Qualification (EQ) of Electrical Components program. Important attributes of a reanalysis include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, and corrective actions (if acceptance criteria are not met).

Under the Hope Creek Environmental Qualification (EQ) of Electric Components program, the reanalysis of an aging evaluation could extend the qualification of the component. If the qualification cannot be extended by reanalysis, the component must be refurbished, replaced, or requalified prior to exceeding the period for which the current qualification remains valid. A reanalysis is to be performed in a timely manner such that sufficient time is available to refurbish, replace, or requalify the component if the reanalysis is unsuccessful. The effects of aging on the intended function(s) will be adequately managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

A.4.5 Loss of Prestress in Concrete Containment Tendons

The Hope Creek containment does not have pre-stressed tendons; therefore this topic is not a TLAA for Hope Creek.

A.4.6 Containment Liner Plate, Metal Containments, and Penetrations Fatigue Analyses

A.4.6.1 Fatigue Analyses of Primary Containment, Attached Piping, and Components

The Hope Creek containment vessel is a Mark I design with a drywell and toroidal suppression chamber. The Hope Creek primary containment was designed in accordance with the ASME Code, Section III. The Mark I analyses are detailed in the Hope Creek Plant Unique Analysis Report (PUAR) and assume 370 multiple SRV lifts and 596 single SRV lifts. Since these analyses include fatigue evaluations based on the occurrence of a limited number of transient cycles during the current licensed term of operation (40 years), they satisfy the requirements of 10 CFR 54.3 and are TLAAs.

Using historical information and projecting the results for 60 years provided a projected number of 592 single SRV lifts for 60 years, and a projected number of 26 multiple lifts for 60 years. Both of these numbers are below the values assumed in the Mark I analyses. Even though the projected number of cycles and the associated cumulative usage factor (CUF) are projected to be within the original design assumptions, all relevant plant transient events will be tracked to ensure that the CUF remains less than 1.0 for all monitored components. Validation for primary containment locations will be performed by monitoring the high CUF limiting containment locations.

All governing fatigue analyses have been reviewed to establish a comprehensive and bounding set of primary containment locations for inclusion in the Metal Fatigue of Reactor Coolant Pressure Boundary aging management program in accordance with 10 CFR 54.21(c)(1)(iii).

A.4.6.2 Primary Containment Process Penetrations and Bellows Fatigue Analysis

The primary containment process piping penetrations use flued heads to connect the process piping to the drywell sleeves. The flued heads for the ASME Section III, Class 1 process piping systems have been analyzed for fatigue. These Class 1 fatigue analyses are based upon thermal cycles specified for the 40-year life for the plant and have therefore been identified as TLAAs requiring evaluation for the period of extended operation. The primary containment process piping penetrations with triple flued heads are anchored to the concrete shield wall around the drywell. As a result, the penetration design includes a flexible bellows to seal the triple flued head to the drywell penetration sleeve. The single and double flued heads are sealed directly to the drywell sleeves, without a flexible bellows. The flexible bellows were designed in accordance with ASME Section III, Class 2 requirements, but this included CUF analyses that have also been identified as TLAA’s requiring evaluation for 60 years. The drywell penetration sleeves were evaluated for cyclic loads under the ASME Section III, Class II requirements, and it was determined that the penetration sleeves were exempt from a detailed fatigue evaluation.

For the Containment Process Penetrations, the maximum 40-year CUF ratio (CUF/allowable) identified for any of these penetrations is 0.957 for feedwater penetrations P-2A and P-2B. All other governing penetration fatigue analyses have been reviewed, and the CUF ratios for all other penetrations are well less than 0.4 (maximum value of 0.057).

The fatigue usage experienced by the flexible bellows was bounded by the corresponding attached triple flued head, therefore, monitoring of the fatigue usage for the triple flued heads will provide assurance that no flexible bellows will exceed its allowable value.

The Metal Fatigue of Reactor Coolant Pressure Boundary Program will manage the effects of aging due to fatigue on the containment process penetrations and the flexible bellows in accordance with 10 CFR 54.21(c)(1)(iii).

A.4.6.3 Vent Line Bellows

Fatigue was evaluated for the vent line bellows that seal the drywell shell to the vent lines, which connect to the torus. The fatigue evaluation for these bellows is based on the number of cycles assumed for the 40-year life of the plant; therefore these analyses satisfy the criteria 10CFR54.3(a) and are evaluated as TLAA in accordance with 10CFR54.21(c).

The maximum differential displacements of the vent line bellows are determined using the results of the torus reanalysis. The displacements at the attachment points of the bellows to the drywell side of the vent line and to the torus side of the vent line are determined for each load case. The differential displacement is computed from these values. The results of each load case are combined to determine the total differential displacements for the controlling load combinations. The results are compared to the allowable bellows displacements. The loads that cause the highest number of displacement cycles at the vent line bellows are seismic loads, SRV loads, and LOCA related loads such as pool swell, condensation oscillation, and chugging. The bellows displacements for these loads are small compared to the maximum allowable displacement and their effect on fatigue is negligible. The thermal loads and internal pressure loads are the largest contributors to bellows displacements. The bellows have a rated capacity of cycles at maximum displacements for normal operating conditions greater than the originally specified (allowable) number of thermal load and internal pressure cycles.

The allowable number of cycles, at maximum displacements for normal operating conditions, are greater than those projected for 60 years, therefore the fatigue evaluation for the vent line bellows remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.4.7 Other Plant–Specific Time Limited Aging Analyses

A.4.7.1 Crane Load Cycle Limit

The load cycle limits for cranes was identified as a potential TLAA. The method of review applicable to the crane cyclic load limit TLAA involves (1) reviewing the existing 40-year design basis to determine the number of load cycles considered in the design of each of the cranes in the scope of license renewal, (2) developing 60-year projections for load cycles for each of the cranes in the scope of license renewal and compare with the number of design cycles for 40 years.

Reactor Building Polar Crane

The Reactor Building Polar Crane at Hope Creek is designed to meet or exceed the design fatigue requirements of the Crane Manufacturers Association of America (CMAA) Specification 70. This evaluation of cycles over the 40-year life is the basis of a safety determination and is therefore a TLAA Analysis.

The Reactor Building Polar Crane is designed in accordance with CMAA Specification 70 for a minimum of 100,000 load cycles. A review of Reactor Building Polar Crane operation during the current life of the plant, including an estimated 200 lifts during original construction, indicates that the total number of lifts above 25 tons to date is less than 800. Using an average rate of 42 lifts per year over the course of 60 years results in the Polar Crane experiencing 2,520 lifts. Additionally, the Polar Crane will be used in the handling of casks, which are projected to be less than 200 lifts from campaign initiation through the period of extended operation. Therefore, the total number of lifts for the Polar Crane has been estimated to be 2,720 for the total life of plant, including the extended period of operation associated with license renewal. This is less than the minimum allowable design value of 100,000 cycles and is therefore acceptable. Therefore, the Reactor Building Polar Crane load cycle fatigue analysis remains valid for 60 years of plant operation in accordance with 10 CFR 54.21(c)(1)(i).

Service Water Intake Structure Gantry Crane

A review of Service Water Intake Structure Gantry Crane operation during the current life of the plant, including an estimated 200 lifts during original construction, indicates that the total number of lifts is less than 600. Using an average rate of 32 lifts per year over the course of 60 years results in the Service Water Intake Structure Gantry Crane experiencing 1,920 lifts. This is less than the minimum allowable design value of 100,000 cycles, and is therefore acceptable. Therefore, the Service Water Intake Structure Gantry Crane load cycle fatigue analysis remains valid for 60 years of plant operation in accordance with 10 CFR 54.21(c)(1)(i).

A.4.7.2 Refueling Bellows Fatigue

A fatigue analysis was performed for each of the two bellows that seal the drywell bulkhead and reactor vessel and the drywell bulkhead and drywell shell for refueling operations. The fatigue analysis for each of these bellows is based on the number of cycles assumed for the 40-year life of the plant; therefore these analyses satisfy the criteria 10 CFR 54.3(a) and are evaluated as TLAA in accordance with 10 CFR 54.21(c).

The bulkhead-to-drywell bellows exhibited the highest calculated fatigue usage. The fatigue usage for both bellows was determined by the number of cycles of startup and shutdown and the number of flood-up events at each refueling outage. The fatigue analysis assumed 360 cycles for startup and shutdown and reflood over the life of the plant. The maximum number of startup and shutdown cycles expected for 60 years of operations at Hope Creek is 180 for startup and shutdown and 55 for refueling operations, for a total of 235 cycles. Since the number of cycles used in the analyses for these events are greater than those projected for 60 years, the fatigue analyses for the refueling bellows remain valid for the period of extended operation.

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

A.4.7.3 Neutron Fluence-Induced Stress Relaxation - Jet Pump Auxiliary Spring Wedges and Slip Joint Clamp

Auxiliary Spring Wedges and a Slip Joint Clamp have been installed on jet pumps at the Hope Creek. Structural analysis of the repair devices evaluated the loss of preload for structural bolting due to exposure to neutron fluence during 40 years of operation. The analysis for each of these devices is based on the fluence assumed for the 40-year life of the device; therefore these analyses satisfy the criteria 10 CFR 54.3(a) and are evaluated as a TLAA in accordance with 10 CFR54.21(c).

Auxiliary Spring Wedges Bolts

Based on an analysis, the auxiliary spring wedge bolts were determined to not experience a fluence level above the analyzed value. Since the fluence bounds that which would be experienced within the period of extended operation, the analysis has been successfully projected for 60 years of plant operation. The current analysis for the auxiliary spring wedges remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

Slip Joint Clamp Bolt

The fluence analysis for the slip joint clamp bolt determined a fluence value equal to that previously evaluated upon reaching 35.4 Effective Full Power Years (EFPY). Two years before reaching the bounding value of 35.4 EFPY an analysis will be performed that demonstrates the function of the component is maintained, or the slip joint clamp will be replaced at a refueling outage before reaching the bounding value of 35.4 EFPY. The slip joint clamp TLAA will be dispositioned in accordance with 10 CFR 54.21(c)(1)(iii) by managing the aging effects by taking the action as described above prior to reaching the bounding value of 35.4 EFPY.

A.5 Hope Creek License Renewal Commitment List

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

* License Renewal Application (LRA)

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA* Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
1	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	Existing program is credited.	A.2.1.1	Ongoing	LRA Section B.2.1.1
2	Water Chemistry	Existing program is credited.	A.2.1.2	Ongoing	LRA Section B.2.1.2
3	Reactor Head Closure Studs	Existing program is credited.	A.2.1.3	Ongoing	LRA Section B.2.1.3
4	BWR Vessel ID Attachment Welds	Existing program is credited.	A.2.1.4	Ongoing	LRA Section B.2.1.4
5	BWR Feedwater Nozzle	Existing program is credited.	A.2.1.5	Ongoing	LRA Section B.2.1.5
6	BWR Control Rod Drive Return Line Nozzle	Existing program is credited.	A.2.1.6	Ongoing	LRA Section B.2.1.6
7	BWR Stress Corrosion Cracking	BWR Stress Corrosion Cracking is an existing program that will be enhanced to include: 1. For the components within the scope of the BWR Stress Corrosion Cracking program, resistant materials will be used for new and replacement components. This includes low carbon stainless piping and stainless	A.2.1.7	Program to be enhanced prior to the period of extended operation.	LRA Section B.2.1.7

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA* Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		steel weld material limited to a maximum carbon content 0.035 wt.% and a minimum ferrite content of 7.5%.			
8	BWR Penetrations	Existing program is credited.	A.2.1.8	Ongoing	LRA Section B.2.1.8
9	BWR Vessel Internals	Existing program is credited. PSEG is committed to implement the BWRVIP guidelines for Hope Creek as follows: • PSEG will inform the NRC staff of any decision to not fully implement a BWRVIP guideline approved by the staff. • PSEG will notify the staff if changes are made to the RPV and its internals' programs that affect the implementation of the BWRVIP guideline. • PSEG will submit any deviation from the existing flaw evaluation guidelines that are specified in the BWRVIP guideline.	A.2.1.9	Ongoing	LRA Section B.2.1.9
10	Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)	Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless steel (CASS) is a new program that will provide for aging management of CASS reactor internal components within the scope of license renewal. The program will include a component specific evaluation of the loss of fracture toughness in accordance with the specified criteria. For those components where loss of fracture toughness may affect function of the component, a supplemental inspection will be performed.	A.2.1.10	Program to be implemented prior to the period of extended operation.	LRA Section B.2.1.10

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA* Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
11	Flow-Accelerated Corrosion	Existing program is credited.	A.2.1.11	Ongoing	LRA Section B.2.1.11
12	Bolting Integrity	Bolting Integrity Program is an existing program that will be enhanced to include: 1. In the following cases, bolting material should not be reused: a. Galvanized bolts and nuts, b. ASTM A490 bolts; and c. Any bolt and nut tightened by the turn of nut method.	A.2.1.12	Program to be enhanced prior to the period of extended operation.	LRA Section B.2.1.12
13	Open-Cycle Cooling Water System	Existing program is credited.	A.2.1.13	Ongoing	LRA Section B.2.1.13
14	Closed-Cycle Cooling Water System	Closed-Cycle Cooling Water System is an existing program that will be enhanced to include: 1. New recurring tasks will be established for enhancing the performance monitoring of the Closed Cycle Cooling Water System. 2. New recurring tasks will be established for enhancing the performance monitoring of the Chilled Water System. 3. A one-time inspection of selected Closed-Cycle Cooling Water program components in stagnant flow areas	A.2.1.14	Program to be enhanced and one-time inspections to be performed prior to the period of extended operation.	LRA Section B.2.1.14

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA* Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		will be conducted to confirm the effectiveness of the Closed-Cycle Cooling Water program. 4. A one-time inspection of selected Closed-Cycle Cooling Water program chemical mixing tanks and associated piping will be conducted to confirm the effectiveness of the Closed-Cycle Cooling Water program on the interior surfaces of the tanks and associated piping. 5. The program will be enhanced such that the plant auxiliary building chilled water system, which is part of the Control Area Chilled Water System, will comply with the pure water control program in accordance with EPRI 1007820 prior to the period of extended operation. 6. A one-time inspection of selected Control Area Chilled Water System components, including the plant auxiliary building chilled water system, will be conducted to confirm the effectiveness of the Closed-Cycle Cooling Water program.			
15	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems is an existing program that will be enhanced to include: 1. Visual inspection of structural components and structural bolts for loss of material due to general,	A.2.1.15	Program to be enhanced prior to the period of extended operation.	LRA Section B.2.1.15

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA* Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		pitting, and crevice corrosion and structural bolting for loss of preload due to self-loosening. 2. Visual inspection of the rails in the rail system for loss of material due to wear. 3. The acceptance criteria will be enhanced to require evaluation of significant loss of material due to corrosion for structural components and structural bolts, and significant loss of material due to wear of rail in the rail system.			
16	Compressed Air Monitoring	Existing program is credited	A.2.1.16	Ongoing	LRA Section B.2.1.16
17	Fire Protection	Fire Protection is an existing program that will be enhanced to include: 1. The routine inspection procedures will be enhanced to provide additional inspection guidance to identify degradation of fire barrier walls, ceilings, and floors for aging effects such as cracking, spalling and loss of material caused by freeze-thaw, chemical attack, and reaction with aggregates. 2. The fire pump supply line functional tests will be enhanced to provide specific guidance for examining	A.2.1.17	Program to be enhanced prior to the period of extended operation.	LRA Section B.2.1.17 Hope Creek Letter LR-N10-0190 RAI B.2.1.17-02

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA* Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		exposed external surfaces of the fire pump diesel fuel oil supply line for corrosion during pump tests. 3. The Halon and Carbon Dioxide fire suppression system functional test procedures will be enhanced to include visual inspection of system piping and component external surfaces for signs of corrosion or other age related degradation, and for mechanical damage. The functional test procedures will also be enhanced to include acceptance criteria stating that identified corrosion or mechanical damage will be evaluated, with corrective action taken as appropriate.			
18	Fire Water System	Fire Water System is an existing program that will be enhanced to include: 1. The Fire Water System aging management program will be enhanced to inspect selected portions of the water based fire protection system piping located aboveground and exposed to the fire water internal environment by non-intrusive volumetric examinations. These inspections shall be performed prior to the period of extended operation and will be performed every 10 years thereafter.	A.2.1.18	Program to be enhanced prior to the period of extended operation. Inspection schedule identified in commitment.	LRA Section B.2.1.18

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA* Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		2. The Fire Water System aging management program will be enhanced to replace or perform 50-year sprinkler head inspections and testing using the guidance of NFPA-25 "Standard for the Inspection, Testing and Maintenance of Water-Based Fire Protection Systems" (2002 Edition), Section 5-3.1.1. These inspections will be performed prior to the 50-year in-service date and every 10-years thereafter.			
19	Aboveground Steel Tanks	Aboveground Steel Tanks is an existing program that will be enhanced to include: 1. The program will be enhanced to include internal UT measurements to measure the wall thickness on the bottom of the tanks supported on a fiber pad on top of the concrete foundation (Fire Water Storage Tanks). Measured wall thickness will be monitored and trended if significant material loss is detected. These thickness measurements of the tank bottom will be taken and evaluated against design thickness and corrosion allowance to ensure that significant degradation is not occurring and the component	A.2.1.19	Program to be enhanced prior to the period of extended operation. Tank bottom UT inspections will also be performed prior to the period of extended operation.	LRA Section B.2.1.19

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA* Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		intended function will be maintained during the extended period of operation. 2. The program will be enhanced to provide routine visual inspections of the carbon steel tanks external surfaces (Fire Water Storage Tanks, Fire Diesel Fuel Oil Tank and 17-Ton CO2 Storage Tank), including removal of tank insulation from the Fire Water Storage tank to detect degradation. These inspections will be performed to detect degraded paint and coatings, and any resulting metal degradation, prior to loss of the tank intended function.			
20	Fuel Oil Chemistry	Fuel Oil Chemistry is an existing program that will be enhanced to include: 1. Equivalent requirements for fuel oil purity and fuel oil testing as described by the Standard Technical Specifications. 2. Addition of biocides, stabilizers and corrosion inhibitors as determined by fuel oil sampling or inspection activities. 3. Internal inspection of Diesel Fire Pump Fuel Oil 280-gallon tank (T-565) using visual inspections and ultrasonic thickness examination of tank bottom. 4. Quarterly water and sediment multilevel sampling on the Diesel Fuel	A.2.1.20	Program to be enhanced and one-time inspections to be performed prior to the period of extended operation.	LRA Section B.2.1.20

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA* Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		Oil Storage Tanks identified in A.2.1.20 5. Internal inspection of the Diesel Fuel Oil Storage Tanks identified in A.2.1.20 using visual inspections and ultrasonic thickness examination of tank bottoms. 6. Quarterly particulate sampling of Diesel Fire Pump Fuel Oil 280-gallon tank (T-565). 7. To confirm the absence of any significant aging effects, a one-time inspection of each of the 550-gallon Diesel Fuel Oil Day Tanks will be performed.			
21	Reactor Vessel Surveillance	Reactor Vessel Surveillance is an existing program that will be enhanced to include: 1. Implement the requirements of BWRVIP-86 Revision 1-A: “BWR Vessel and Internals Project Updated BWR Integrated Surveillance Program (ISP) Implementation Plan”, October 2012. 2. If future plant operations exceed the limitations specified in RG 1.99, the impact of plant operation changes on the extent of reactor vessel embrittlement will be evaluated and the NRC will be notified. Similarly, if future plant operation exceeds the bounds established by surveillance	A.2.1.21	Program to be enhanced prior to the period of extended operation.	LRA Section B.2.1.21

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA* Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		data that are to determine Upper Shelf Energy or P-T limits, then the impact of plant operation changes on the extent of reactor vessel embrittlement will be evaluated and the NRC will be notified. Additionally, when all the surveillance capsules are removed, then operating restrictions will be established to ensure that the plant is operated within the conditions to which the surveillance capsules were exposed. If the reactor vessel exposure conditions (neutron flux, spectrum, irradiation temperature, etc.) are altered, then the basis for the projection to 60 years is reviewed; and, if deemed appropriate, a revised fluence projection is prepared and the effects of the revised fluence analysis on neutron embrittlement calculations will be evaluated. If necessary an active surveillance program will be re-instituted for Hope Creek. The employment of additional surveillance specimens will be coordinated through the BWRVIP Integrated Surveillance Program (ISP). Any changes to the reactor vessel exposure conditions and the potential need to re-institute a vessel surveillance program will be discussed with the NRC staff prior to changing the plant's licensing basis.			

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA* Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
22	One-Time Inspection	One-Time Inspection is a new program and will be used for the following: 1. To confirm the effectiveness of the Water Chemistry program to manage the loss of material, cracking, and the reduction of heat transfer aging effects for aluminum, copper alloy, ductile cast iron, gray cast iron, nickel alloy, steel, stainless steel, and cast austenitic stainless steel in treated water, steam, sodium pentaborate and reactor coolant environments. 2. To confirm the effectiveness of the Fuel Oil Chemistry program to manage the loss of material aging effect for copper alloy, steel, galvanized steel and stainless steel in a fuel oil environment. 3. To confirm the effectiveness of the Lubricating Oil Analysis program to manage the loss of material and the reduction of heat transfer aging effects for copper alloy, gray cast iron, steel and stainless steel in a lubricating oil environment. 4. To confirm loss of material in carbon steel piping and fittings is insignificant in an air/gas-wetted (internal) environment. The sample plan for inspections associated with the One-Time Inspection program will be developed to ensure there are adequate	A.2.1.22	Program to be implemented prior to the period of extended operation. One-time inspections to be performed within the ten-year period prior to the period of extended operation.	LRA Section B.2.1.22 Hope Creek Letter LR-N11-0006 RAI B.2.1.22-1

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA* Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		inspections to address each of the material, environment, and aging effect combinations. A sample size of 20% of the population (up to a maximum of 25 inspections) will be established for each of the sample groups.			
23	Selective Leaching of Materials	Selective Leaching of Materials is a new program that will include one-time inspections of a representative sample of susceptible components to determine where loss of material due to selective leaching is occurring. A sample size of 20% of susceptible components will be subjected to a one-time inspection with a maximum of 25 inspections for each of the susceptible material groups. Where selective leaching is identified, further aging management activities will be implemented such that the component intended function is maintained consistent with the current licensing basis through the period of extended operation.	A.2.1.23	Program to be implemented prior to the period of extended operation. One-time inspections to be performed within the ten-year period prior to the period of extended operation.	LRA Section B.2.1.23 Hope Creek Letter LR-N10-0319 LRA Supplement Hope Creek Letter LR-N11-0006 RAI B.2.1.23-1
24	Buried Piping Inspection	Buried Piping Inspection is an existing program that will be enhanced to include: 1. At least one (1) opportunistic or focused excavation and inspection will be performed on each of the material groupings, which include carbon steel, ductile cast iron, and gray cast iron piping and components during each ten (10) year period, beginning ten (10) years prior to entry into the period of extended operation. A second opportunistic or focused excavation and inspection on a	A.2.1.24	Program to be enhanced prior to the period of extended operation. Inspection schedule identified in commitment.	LRA Section B.2.1.24 Hope Creek Letter LR-N10-0323 RAI B.2.1.24 Hope Creek Letter LR-N10-0371 RAI B.2.1.24-02

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA* Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		carbon steel piping segment, which is not cathodically protected, will be performed on the Service Water System during each ten year period, beginning ten years prior to entry into the period of extended operation. A different segment will be inspected in each ten year period.			
25	External Surfaces Monitoring	External Surfaces Monitoring is a new program that directs visual inspections of components such as piping, piping components, ducting and other components in the scope of license renewal, exposed to an air environment, to manage aging effects.	A.2.1.25	Program to be implemented prior to the period of extended operation.	LRA Section B.2.1.25
26	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components is a new program that manages the aging of the internal surfaces of piping, piping components and piping elements, tanks and ducting components.	A.2.1.26	Program to be implemented prior to the period of extended operation.	LRA Section B.2.1.26
27	Lubricating Oil Analysis	Existing program is credited.	A.2.1.27	Ongoing	LRA Section B.2.1.27
28	ASME Section XI, Subsection IWE	ASME Section XI, Subsection IWE is an existing program that will be enhanced to include: 1. Install an internal moisture barrier at the junction of the drywell concrete floor and the steel drywell shell prior to the period of extended operation. 2. Require inspection of the moisture barrier for loss of sealing in accordance with IWE 2500 after it is	A.2.1.28	Program to be enhanced prior to the period of extended operation. Inspection schedule identified in commitment.	LRA Section B.2.1.28 Hope Creek Letter LR-N10-0190 RAI B.2.1.28-01

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA* Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		installed. 3. Verify that the reactor cavity seal rupture drain lines are clear from blockage and that the monitoring instrumentation is functioning properly once prior to the period of extended operation, and one additional time during the first ten years of the period of extended operation. 4. Establish drainage capability from the bottom of the drywell air gap on or before June 30, 2015. The drywell air gap will be divided into four approximately equal quadrants. Drainage consists of one drain in each quadrant for a total of four drains. Each drain will be open at the bottom of the drywell air gap and be capable of draining water from the air gap. Verify that drains at the bottom of the drywell air gap are clear from blockage once prior to the period of extended operation, and one additional time during the first ten years of the period of extended operation. 5. Investigate the source of any leakage detected by the reactor cavity seal rupture drain line instrumentation and assess its impact on the drywell shell.			Hope Creek Letter LR-N10-0291 RAI B.2.1.28-01 Hope Creek Letter LR-N11-0016 RAI B.2.1.28-3 Hope Creek Letter LR-N11-0164 RAI B.2.1.28-4

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA* Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		6. After drainage has been established from the bottom of the air gap from all four drains, monitor the drains at the bottom of the drywell air gap daily for leakage in the event leakage is detected by the reactor cavity seal rupture drain line instrumentation. 7. Monitor penetration sleeve J13 daily for water leakage when the reactor cavity is flooded up. In addition, perform a walkdown of the torus room to detect any leakage from other drywell penetrations. These actions shall continue until corrective actions are taken to prevent leakage through J13 or through the four air gap drains. 8. Until drainage is established from all four drains, when the reactor cavity is flooded up, perform boroscope examination of the bottom of the drywell air gap through penetrations located at elevation 93’ in four quadrants, 90 degrees apart. The personnel performing the boroscope examination shall be certified as VT-1 inspectors in accordance ASME Section XI, Subsection IWA-2300, requirements. The examiners will look for signs of water accumulation and drywell shell corrosion. Adverse conditions will be documented and addressed in the corrective action program.			

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA* Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		After drainage has been established from the bottom of the air gap from all four drains, monitor the lower drywell air gap drains daily for water leakage when the reactor cavity is flooded up. 9. Until drainage is established from all four drains, perform UT thickness measurements each refuel outage from inside the drywell in the area of the drywell shell below the J13 penetration sleeve area to determine if there is a significant corrosion rate occurring in this area due to periodic exposure to reactor cavity leakage. In addition, UT measurement shall be performed each refuel outage around the full 360 degree circumference of the drywell between elevations 86'-11" and 88'-0" (underside of the torus down comer vent piping penetrations). Inspection and acceptance criteria will be in accordance with IWE-2000 and IWE-3000 respectively. The results of the UT measurements shall be used to establish a corrosion rate and demonstrate that the effects of aging will be adequately managed such that the drywell can perform its intended function until April 11, 2046. Evidence of drywell shell degradation will be documented and addressed in the corrective action program.			

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA* Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		After drainage has been established from the bottom of the air gap from all four drains, UT thickness measurements will be taken each of the next three refueling outages at the same locations as those previously examined as described above. These UT thickness measurements will be compared to the results of the previous UT inspections and, if corrosion is ongoing, a corrosion rate will be determined for the dry well shell. In the event a significant corrosion rate is detected, the condition will be entered in the corrective action process for evaluation and extent of condition determination. 10. The cause of the reactor cavity water leakage will be investigated and repaired, if practical, before PEO. If repairs cannot be made prior to the PEO, the program will be enhanced to incorporate the following aging management activities, as recommended in the Final Interim Staff Guidance LR-ISG-2006-01. a. Identify drywell surfaces requiring examination and implement augmented inspections for the period of extended operation in accordance with IWE-1240, as identified in Table IWE-2500-1,			

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA* Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		Examination Category E-C. b. Demonstrate through the use of augmented inspections that corrosion is not occurring or that corrosion is progressing so slowly that the age-related degradation will not jeopardize the intended function of the drywell shell through the period of extended operation. c. Develop a corrosion rate that can be inferred from past UT examinations. If degradation has occurred, evaluate the drywell shell using the developed corrosion rate to demonstrate that the drywell shell will have sufficient wall thickness to perform its intended function through the period of extended operation.			
29	ASME Section XI, Subsection IWF	Existing program is credited.	A.2.1.29	Ongoing	LRA Section B.2.1.29
30	10 CFR Part 50, Appendix J	Existing program is credited.	A.2.1.30	Ongoing	LRA Section B.2.1.30
31	Masonry Wall Program	Masonry Wall is an existing program that will be enhanced to include: 1. Additional buildings and masonry walls as described in A.2.1.31. 2. Add an Examination Checklist for	A.2.1.31	Program to be enhanced prior to the period of extended operation.	LRA Section B.2.1.31

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA* Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		masonry wall inspection requirements. 3. Specify an inspection frequency of not greater than 5 years for masonry walls.			
32	Structures Monitoring Program	Structures Monitoring is an existing program that will be enhanced to include: 1. Additional structures and components as described in A.2.1.32 2. Concrete structures will be observed for a reduction in equipment anchor capacity due to local concrete degradation. This will be accomplished by visual inspection of concrete surfaces around anchors for cracking, and spalling. 3. Clarify inspection criteria for loss of material due to corrosion and pitting of additional steel components, such as embedments, panels and enclosures, doors, siding, metal deck, and anchors. 4. Perform a one-time inspection of the external stainless steel surfaces of the expansion bellows at Condensate Storage Tank Dike for loss of material due to corrosion, within the ten-year period prior to the period of extended operation. 5. Require inspection of penetration seals, structural seals and elastomers for degradations that will lead to a	A.2.1.32	Program to be enhanced prior to the period of extended operation. One-time inspection to be performed within the ten-year period prior to the period of extended operation.	LRA Section B.2.1.32

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA* Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		loss of sealing by visual inspection of the seal for hardening, shrinkage and loss of strength. 6. Require monitoring of vibration isolators, associated with component supports other than those covered by ASME XI, Subsection IWF. 7. Add an Examination Checklist for masonry wall inspection requirements. 8. Parameters monitored for wooden components will be enhanced to include: Change in Material Properties, Loss of Material due to Insect Damage and Moisture Damage. 9. Specify an inspection frequency of not greater than 5 years for structures including submerged portions of the Service Water Intake Structure. 10. Require individuals responsible for inspections and assessments for structures to have a B.S. Engineering degree and/or Professional Engineer license, and a minimum of four years experience working on building structures. 11. Perform periodic sampling, testing, and analysis of ground water chemistry for pH, chlorides, and sulfates on a frequency of 5 years.			

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA* Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		12. Require supplemental inspections of the in scope structures within 30 days following extreme environmental or natural phenomena (large floods, significant earthquakes, hurricanes, and tornadoes). 13. Perform a chemical analysis of ground or surface water in-leakage when there is significant in-leakage or there is reason to believe that the in-leakage may be damaging concrete elements or reinforcing steel. 14. Implementing procedures will be enhanced to include additional acceptance criteria details specified in ACI 349.3R-96.			
33	RG 1.127, Inspection of Water-Control Structures	RG 1.127, Inspection of Water-Control Structures Associated With Nuclear Power Plants is an existing program that will be enhanced to include: 1. Shoreline Protection and Dike structures will be added to the program. 2. Parameters monitored for wooden components will be enhanced to include change in material properties and loss of material due to insect damage and moisture damage. 3. The inspection requirement for submerged concrete structural	A.2.1.33	Program to be enhanced prior to the period of extended operation.	LRA Section B.2.1.33

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA* Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		components will be enhanced to require that inspections be performed by dewatering a pump bay or by a diver if the pump bay is not dewatered. 4. Specify an inspection frequency of not greater than 5 years for structures including submerged portions of the Service Water Intake Structure. 5. Require supplemental inspections of the in scope structures within 30 days following extreme environmental or natural phenomena (large floods, significant earthquakes, hurricanes, and tornadoes).			
34	Protective Coating Monitoring and Maintenance Program	Existing program is credited.	A.2.1.34	Ongoing	LRA Section B.2.1.34
35	Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements is a new program and will be used to manage aging of non-EQ cables and connections during the period of extended operation.	A.2.1.35	Program and initial inspections to be implemented prior to the period of extended operation.	LRA Section B.2.1.35
36	Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits	Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits is a new program that will be implemented to manage the aging of the cable and connection insulation of the in scope portions of the Leak Detection and Radiation Monitoring System, and the Neutron Monitoring System.	A.2.1.36	Program and initial assessment of testing and calibration results to be implemented prior to the period of extended operation.	LRA Section B.2.1.36

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA* Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
37	Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements is a new program that will be used to manage the aging effects and mechanisms of non-EQ, in scope inaccessible power cables (480V, 4,160V, 13,800V). The cable test frequency will be established based on test results and industry operating experience. The maximum time between tests will be no longer than 6 years. Manholes and cable vaults associated with the cables included in this aging management program will be inspected for water collection (with water removal as necessary) with the objective of minimizing the exposure of power cables to significant moisture. Prior to the period of extended operation, the frequency of inspections for accumulated water will be established based on inspection results to minimize the exposure of power cables to significant moisture. The maximum time between inspections will be no longer than one year. The Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements aging management program will be enhanced as follows: 1. Add low voltage power cables (480 volts or greater) to the scope of the program	A.2.1.37	Enhanced Program and initial cable tests and manhole inspections to be implemented prior to the period of extended operation. Test and inspection schedule identified in commitment.	LRA Section B.2.1.37 Hope Creek Letter LR-N10-0190 RAI B.2.1.37-01 Hope Creek Letter LR-N10-0190 RAI B.2.1.37-02 Hope Creek Letter LR-N10-0325 LRA Supplement Hope Creek Letter LR-N10-0360 LRA Supplement

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA* Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		2. Change cable testing maximum frequency from 10 years to 6 years. Change cable vault and manhole inspection maximum frequency from 2 years to 1 year.			
38	Metal-Enclosed Bus	Metal Enclosed Bus is a new program that will manage the aging of in-scope metal enclosed busses.	A.2.1.38	Program and initial inspections to be implemented prior to the period of extended operation.	LRA Section B.2.1.38
39	Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements is a new program that will be used to confirm the absence of an aging effect with respect to electrical cable connection stressors. A representative sample of non-EQ electrical cable connections will be selected, for one-time testing considering application (medium and low voltage), circuit loading (high loading) and location, with respect to connection stressors.	A.2.1.39	Program and one-time testing to be implemented prior to the period of extended operation.	LRA Section B.2.1.39
40	High Voltage Insulators	High Voltage Insulators is a new program that manages the degradation of insulator quality due to the presence of salt deposits or surface contamination.	A.2.2.1	Program to be implemented prior to the period of extended operation.	LRA Section B.2.2.1

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA* Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
41	Periodic Inspection	Periodic Inspection is a new program that manages the aging of piping, piping components, piping elements, ducting components, tanks and heat exchanger components.	A.2.2.2	Program to be implemented prior to the period of extended operation.	LRA Section B.2.2.2
42	Aboveground Non-Steel Tanks	Aboveground Non-Steel Tanks is a new program that will manage loss of material of outdoor non-steel tanks. The Aboveground Non-Steel Tanks program will include a UT wall thickness inspection of the bottom of the only tank in the program, which is the stainless steel condensate storage tank. The UT measurements will be taken to ensure that significant degradation is not occurring and that the component intended function will be maintained during the extended period of operation.	A.2.2.3	Program to be implemented prior to the period of extended operation. Tank bottom UT inspections will also be performed prior to the period of extended operation.	LRA Section B.2.2.3
43	Buried Non-Steel Piping Inspection	Buried Non-Steel Piping Inspection is an existing program that will be enhanced to include: 1. At least one (1) opportunistic or focused excavation and inspection will be performed on buried reinforced concrete piping and components during each ten (10) year period, beginning ten (10) years prior to entry into the period of extended operation. 2. At least one (1) opportunistic or focused excavation and inspection will be performed on Condensate Storage and Transfer System buried	A.2.2.4	Program to be enhanced prior to the period of extended operation. Inspection schedule identified in commitment.	LRA Section B.2.2.4 Hope Creek Letter LR-N10-0323 RAI B.2.1.24 Hope Creek Letter LR-N10-0371 RAI B.2.1.24-02

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA* Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
		stainless steel piping and components, which contain fluid that exceed EPA drinking water limits, during each ten (10) year period, beginning ten (10) years prior to entry into the period of extended operation. 3. Guidance for inspection of concrete aging effects.			
44	Boral Monitoring Program	Boral Monitoring Program is an existing program that will be enhanced to include: 1. Inspection, testing and evaluation of one coupon from the Hope Creek spent fuel pool prior to the period of extended operation and one coupon within the first 10 years after entering the period of extended operation. Testing will include dimensional and neutron attenuation measurements with an acceptance criteria of no more than a 10% increase in thickness and no more than a 5% decrease in B-10 areal density.	A.2.2.5	Program to be enhanced prior to the period of extended operation. Inspection schedule identified in commitment.	LRA Section B.2.2.5 Hope Creek Letter LR-N10-0154 RAI 2.2.5-1
45	Small-Bore Class 1 Piping Inspection	Small-Bore Class 1 Piping is a new program that will manage the aging effects of cracking in small-bore (greater than or equal to NPS 1 and less than NPS 4) Class 1 piping through the use of a combination of volumetric examinations and visual inspections.	A.2.2.6	Program to be implemented prior to the period of extended operation, with the supplemental inspections performed within the six year period prior to the period of extended operation.	LRA Section B.2.2.6 Hope Creek Letter LR-N10-0415 RAI B.2.2.6-01

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA* Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
46	Metal Fatigue of the Reactor Coolant Pressure Boundary	Metal Fatigue of the Reactor Coolant Pressure Boundary is an existing program that will be enhanced to include: 1. Adding transients beyond those defined in the Technical Specifications and the UFSAR, and expanding the fatigue monitoring program to encompass other components identified to have fatigue as an analyzed aging effect, which require monitoring. 2. Using a software program to automatically count transients and calculate cumulative usage on select components. At this time only cycle based fatigue monitoring will be used. If stress based fatigue monitoring is used in the future, it will consider the six stress terms in accordance with the methodology from ASME Section III, Subsection NB, Subarticle NB-3200. 3. Addressing the effects of the reactor coolant environment on component fatigue life by assessing the impact of the reactor coolant environment on a sample of critical components for the plant identified in NUREG/CR-6260. 4. Requiring a review of additional reactor coolant pressure boundary locations if the usage factor for one of the environmental fatigue sample locations approaches its design limit	A.3.1.1	Program to be enhanced prior to the period of extended operation.	LRA Section B.3.1.1 Hope Creek Letter LR-N10-0356 RAI 4.3-01

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA* Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
47	Environmental Qualification of Electric Components (EQ)	Existing program is credited.	A.3.1.2	Ongoing	LRA Section B.3.1.2
48	New P-T Curves	Revised Pressure-Temperature (P-T) limits will be submitted to the NRC when necessary to comply with 10 CFR 50 Appendix G.	A.4.2.3	Ongoing	LRA Section 4.2.3
49	RPV Circumferential Weld Examination Relief	PSEG will request relief from the requirement to perform volumetric examinations of the Reactor Pressure Vessel Circumferential Welds, in accordance with 10 CFR 50.55(a)	A.4.2.4	Prior to the period of extended operation.	LRA Section 4.2.4
50	Operating Experience Review	PSEG will perform an evaluation of operating experience at extended power uprate (EPU) levels prior to the period of extended operation to ensure that operating experience at EPU levels is properly addressed by the aging management programs. The evaluation will include Hope Creek and other BWR plants operating at EPU levels.	All programs	Prior to the period of extended operation.	NUREG-1800 Section 3.0.2
51	Reactor Internals Components – Core Plate Rim Hold-Down Bolts	PSEG will perform one of the following: 1. Install core plate wedges, or 2. Perform an analysis that demonstrates the component function is maintained.	A.4.2.7	Prior to the period of extended operation.	LRA Section 4.2.7 and Appendix C

NO.	PROGRAM OR TOPIC	COMMITMENT	UFSAR SUPPLEMENT LOCATION (LRA* Section)	ENHANCEMENT OR IMPLEMENTATION SCHEDULE	SOURCE
52	Jet Pump Slip Joint Clamp Bolt	PSEG will replace the slip joint clamp or perform an analysis that demonstrates the function of the component is maintained.	A.4.7.3	Two years before reaching the bounding value of 35.4 EFPY, perform the analysis, or replace the slip joint clamp at a refueling outage prior to reaching the bounding value of 35.4 EFPY	LRA Section 4.7.3 Hope Creek Letter LR-N10-0344
53	Metal Fatigue of Reactor Coolant Pressure Boundary	Environmental fatigue calculations for the Hope Creek Alloy 600 locations will use the data and the methodology that is described in NUREG/CR-6909 or later revisions/reports for Ni-Cr-Fe alloys in the determination of the F_{en} factor and fatigue usage.	A.3.1.1 A.4.3.5	Upon calculation revision/update	Hope Creek Letter LR-N10-0344 RAI 4.3-07
54	Metal Fatigue of Reactor Coolant Pressure Boundary	PSEG will perform a review of design basis ASME Code Class 1 fatigue evaluations to determine whether the NUREG/CR-6260 based locations that have been evaluated for the effects of the reactor coolant environment on fatigue usage are the limiting locations for the Hope Creek plant configuration. If more limiting locations are identified, the most limiting location will be evaluated for the effects of the reactor coolant environment on fatigue usage. If any of the limiting locations consist of nickel alloy, NUREG/CR-6909 methodology for nickel alloy will be used in the evaluation.	A.3.1.1 A.4.3.5	Prior to the period of extended operation.	Hope Creek Letter LR-N10-0440 Confirmatory Item 4.3.5.2-1