3.14 LICENSE RENEWAL PROGRAMS, TLAA, AND COMMITMENTS

3.14.1 INTRODUCTION

The License Renewal Rule, 10 CFR Part 54, governs the issuance of renewed operating licenses for nuclear power plants. The renewed SSES operating licenses were granted on November, 24, 2009 to be effective until July 17, 2042 for Unit 1 and March 23, 2044 for Unit 2.

This section contains the license renewal information to be included in the Final Safety Analysis Report as required by 10 CFR 54.21(d). The Safety Evaluation Report (SER) NUREG-1931, for renewed operating licenses for SSES Units 1 & 2 contains the results of the NRC review of technical information required by 10 CFR 54.21(a) and (c) as submitted in the SSES License Renewal Application (LRA) and supplemental RAI responses. The programs and activities that will be implemented to manage the effects of aging for the period of extended operation are listed throughout NUREG-1931. In addition, this section contains a listing of commitments associated with license renewal as identified in Appendix A of NUREG-1931.

3.14.2 AGING MANAGEMENT PROGRAMS (AMP)

The license renewal integrated plant assessment identified existing and new aging management programs necessary to provide reasonable assurance that 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. This section describes the required aging management programs identified during the integrated plant assessment. Except for one-time inspections, these programs will be implemented during the period of extended operation. One-time inspections will be conducted within the 10-year period prior to beginning the period of extended operation.

Three elements of an effective aging management program that are common to all aging management programs are corrective actions, confirmation process, and administrative controls. These elements are included in the SSES Operational Quality Assurance (OQA) Program, which implements the requirements of 10 CFR 50, Appendix B. Prior to the period of extended operation, the elements of corrective action, confirmation process, and administrative controls in the SSES OQA Program will be applied to required aging management programs for both safety-related and nonsafety-related structures and components determined to require aging management during the period of extended operation.

3.14.2.1 Area-Based NSAS Inspection (NUREG 1931, Section 3.0.3.3.1)

The Area-Based NSAS Inspection detects and characterizes the conditions on the internal surfaces of nonsafety-related components exposed to non-radioactive equipment/area drainage water or potable water environments. The Area-Based NSAS Inspection also detects and characterizes specific conditions on the internal surface of copper alloys exposed to raw water from the spray pond/cooling tower. Components identified as non-safety affecting safety (NSAS) are those nonsafety-related components with the potential to prevent a safety-related system or component from performing its safety function. The conditions in these environments are not expected to result in sufficient degradation to cause spatial interaction with safety-related components or are expected to result in effects that progress very slowly. To ensure that spatial interactions do not occur that impair or prevent a safety-related function, a focused characterization of conditions is performed to provide confirmation of a lack of degradation or to serve as the basis for recurring actions during the period of extended operation, if required.

The Area-Based NSAS Inspection is a new one-time inspection that will be implemented prior to the period of extended operation. The inspection activities will be conducted within the 10-year period prior to the period of extended operation.

3.14.2.2 Bolting Integrity Program (NUREG 1931, Section 3.0.3.2.5)

The Bolting Integrity Program is a combination of existing SSES activities that, in conjunction with other credited programs, addresses the management of aging for the bolting of subject mechanical components within the scope of license renewal. The Bolting Integrity Program relies on manufacturer/vendor information and industry recommendations for the proper selection, assembly and maintenance of bolting for pressure-retaining closures. The Bolting Integrity Program includes, through the Inservice Inspection (ISI) Program and System Walkdown Program, the periodic inspection of bolting for indication of degradation such as leakage, loss of material due corrosion, or cracking.

Prior to the period of extended operation, the Bolting Integrity Program will be enhanced to include a specific precaution against the use of sulfur (sulfide) containing compounds as a lubricant for bolted connections.

3.14.2.3 Buried Piping and Tanks Inspection Program (NUREG 1931, Section 3.0.3.2.13)

The Buried Piping and Tanks Inspection Program manages the effects of corrosion on the external surfaces of piping and tanks exposed to a buried environment. The Buried Piping and Tanks Inspection Program will be a combination of a prevention program (consisting of protective coatings and wrappings, where appropriate) and a condition monitoring program (consisting of visual inspections).

The Buried Piping and Tanks Inspection Program is a new aging management program that will be implemented prior to the period of extended operation.

3.14.2.4 Buried Piping Surveillance Program (NUREG 1931, Section 3.0.3.2.10)

The Buried Piping Surveillance Program manages the effects of corrosion on the external surfaces of piping with damaged coatings exposed to a buried environment. The Buried Piping Surveillance Program will be a combination of a prevention program (consisting of cathodic protection) and a condition monitoring program (consisting of periodic testing).

The Buried Piping Surveillance Program is a new aging management program that will be implemented prior to the period of extended operation.

3.14.2.5 BWR CRD Return Line Nozzle Program (NUREG 1931, Section 3.0.3.2.2)

The BWR CRD Return Line Nozzle Program is an existing program that manages cracking of the control rod drive return line (CRDRL) nozzle, cap, and connecting weld. The program was developed in response to industry events involving the CRD return line nozzle.
SSES modified the CRD System by cutting the CRDRL and capping the nozzle prior to initial plant startup. The CRDRL was not rerouted. SSES has completed all of the requirements

specified in NUREG-0619 for the CRD System modifications performed at SSES, including the final liquid penetrant testing (PT) inspection and system performance testing. The SSES BWR CRD Return Line Nozzle Program monitors the effects of cracking on the intended function of the CRDRL nozzle by performing inservice inspections in conformance with the ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWB, Table IWB 2500-1 (edition and addenda described in 3.14.2.23). Any cracks that are detected will be dispositioned in accordance with the requirements of ASME Section XI.

3.14.2.6 BWR Feedwater Nozzle Program (NUREG 1931, Section 3.0.3.1.4)

The BWR Feedwater Nozzle Program is an existing program that manages cracking of the feedwater nozzles.

The program includes (a) enhanced inservice inspection in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWB, Table IWB 2500-1 (edition and addenda described in 3.14.2.23) and the recommendations of report GE-NE-523-A71-0594, and (b) system modifications (completed on the spargers prior to initial startup) to mitigate cracking. The program specifies periodic ultrasonic inspection of critical regions of the feedwater nozzles.

3.14.2.7 BWR Penetrations Program (NUREG 1931, Section 3.0.3.2.3)

The BWR Penetrations Program is an existing program that manages cracking of selected reactor vessel penetrations. The BWR Penetrations Program is implemented via the Inservice Inspection (ISI) Program in compliance with ASME Section XI and the Boiling Water Reactor Vessel and Internals Project (BWRVIP) guidelines.

The program includes (a) inspection and flaw evaluation in conformance with the guidelines of NRC-approved BWRVIP reports and BWRVIP-27-A, BWRVIP-47-A, BWRVIP-49-A, and BWRVIP-74-A and (b) monitoring and control of reactor coolant water chemistry in accordance with the guidelines of the latest revision of BWRVIP for “BWR Water Chemistry Guidelines” to ensure the long-term integrity and safe operation of reactor vessel internal components. The BWRVIP-27-A report addresses the standby liquid control system nozzle or housing, the BWRVIP-47-A report addresses the control rod drive and flux monitor penetrations in the lower plenum, the BWRVIP-49-A report provides guidelines for instrument penetrations, and the BWRVIP-74-A report addresses the reactor vessel flange leak off penetrations and the reactor vessel drain penetrations.

3.14.2.8 BWR Stress Corrosion Cracking (SCC) Program (NUREG 1931, Section 3.0.3.1.5)

The BWR Stress Corrosion Cracking (SCC) Program is an existing program that manages stress corrosion cracking for stainless steel piping, valves, flow instruments, and pump casings. The program to manage stress corrosion cracking in pressure boundary piping made of stainless steel is delineated in NUREG-0313, Revision 2, and NRC Generic Letter 88-01 and its Supplement 1.

The program includes (a) preventive measures to mitigate intergranular stress corrosion cracking (IGSCC), and (b) inspection and flaw evaluation to monitor IGSCC and its effects. The NRC-approved report BWRVIP-75-A allows for modifications of inspection scope in the Generic Letter 88-01 program.

3.14.2.9 BWR Vessel ID Attachment Welds Program (NUREG 1931, Section 3.0.3.1.3)

The BWR Vessel ID Attachment Welds Program is an existing program that manages cracking of the welds for internal attachments to the reactor pressure vessel.

The program includes (a) inspection and flaw evaluation in accordance with the guidelines of the NRC-approved report BWRVIP-48-A, and (b) monitoring and control of reactor coolant water chemistry in accordance with the guidelines of the latest revision of BWRVIP for “BWR Water Chemistry Guidelines” to ensure the long-term integrity and safe operation of the vessel inside diameter (ID) attachment welds. The BWR Vessel ID Attachment Welds Program is based on the inspection and flaw evaluation guidelines of the BWRVIP, and is implemented by the Inservice Inspection (ISI) Program in accordance with the ASME Code, Section XI, Table IWB 2500-1.

3.14.2.10 BWR Vessel Internals Program (NUREG 1931, Section 3.0.3.2.4)

The BWR Vessel Internals Program is an existing program that manages aging of the reactor vessel internals in accordance with the requirements of ASME Section XI and the BWRVIP documents. The purpose of the BWR Vessel Internals Program is to manage cracking, loss of material, and reduction of fracture toughness for various subcomponents of the reactor vessel internals.

The program includes (a) inspection and flaw evaluation in conformance with the guidelines of applicable and NRC-approved BWRVIP reports, and (b) monitoring and control of reactor coolant water chemistry in accordance with the guidelines of the latest revision of BWRVIP for “BWR Water Chemistry Guidelines” to ensure the long-term integrity and safe operation of reactor vessel internal components.

3.14.2.11 BWR Water Chemistry Program (NUREG 1931, Section 3.0.3.1.1)

The BWR Water Chemistry Program is an existing program that mitigates damage due to loss of material and cracking of plant components that are within the scope of license renewal and contain treated water. The program manages the relevant conditions that could lead to the onset and propagation of a loss of material or cracking through proper monitoring and control consistent with pertinent EPRI water chemistry guidelines and the latest revision of BWRVIP for “BWR Water Chemistry Guidelines”. The relevant conditions are specific parameters such as sulfates, halogens, dissolved oxygen, and conductivity that could lead to or are indicative of conditions for, corrosion, erosion, or stress corrosion cracking (SCC) of susceptible materials. The BWR Water Chemistry Program is a mitigation program.

The BWR Water Chemistry Program is supplemented by the Chemistry Program Effectiveness Inspection which provides verification of the effectiveness of the BWR Water Chemistry Program in mitigating the effects of aging.

3.14.2.12 Chemistry Program Effectiveness Inspection (NUREG 1931, Section 3.0.3.1.10)

The Chemistry Program Effectiveness Inspection detects and characterizes the condition of materials in representative low flow and stagnant areas of plant systems influenced by the BWR Water Chemistry Program, the Closed Cooling Water Chemistry Program, and the Fuel Oil Chemistry Program. The inspection provides direct evidence as to whether, and to what extent, cracking or a loss of material has occurred. The Chemistry Program Effectiveness Inspection will provide confirmation of the effectiveness of the BWR Water, Closed Cooling Water, and Fuel Oil Chemistry Programs in managing the effects of aging.

The Chemistry Program Effectiveness Inspection is a new one-time inspection that will be implemented prior to the period of extended operation. The inspection activities will be conducted within the 10-year period prior to the period of extended operation.

3.14.2.13 Closed Cooling Water Chemistry Program (NUREG 1931, Section 3.0.3.2.7)

The Closed Cooling Water Chemistry Program is an existing program that mitigates damage due to loss of material and cracking of plant components that are within the scope of license renewal and that contain treated water in a closed cooling water system or component (e.g., a heat exchanger) served by a closed cooling water system. The program manages the relevant conditions that could lead to the onset and propagation of a loss of material or cracking through proper monitoring and control of corrosion inhibitor concentrations consistent with the pertinent EPRI water chemistry guideline. The Closed Cooling Water Chemistry Program is a mitigation program.

The Closed Cooling Water Chemistry Program is supplemented by the Chemistry Program Effectiveness Inspection which provides verification of the effectiveness of the Closed Cooling Water Chemistry Program in mitigating the effects of aging.

3.14.2.14 Condensate and Refueling Water Storage Tanks Inspection (NUREG 1931, Section 3.0.3.1.9)

The Condensate and Refueling Water Storage Tanks Inspection detects and characterizes the conditions on the bottom surfaces of the Condensate Storage Tanks and the Refueling Water Storage Tank. The inspection provides direct evidence through volumetric and/or visual examination as to whether, and to what extent, a loss of material due to crevice, general or pitting corrosion has occurred or is likely to occur in inaccessible areas (i.e., tank base/bottom) that could result in a loss of intended function.

The Condensate and Refueling Water Storage Tanks Inspection is a new one-time inspection that will be implemented prior to the period of extended operation. The inspection activities will be conducted within the 10-year period prior to the period of extended operation.

3.14.2.15 Containment Leakage Rate Test Program (NUREG 1931, Section 3.0.3.1.22)

The Containment Leakage Rate Test Program is an existing program that manages aging effects for the Primary Containment and systems penetrating the Primary Containment, which are the containment liner and Primary Containment penetrations including personnel airlock, equipment hatches, and control rod drive hatch. The Containment Leakage Rate Test Program provides assurance that leakage from the Primary Containment will not exceed maximum values for containment leakage.

3.14.2.16 Cooling Units Inspection (NUREG 1931, Section 3.0.3.1.11)

The Cooling Units Inspection detects and characterizes the condition of aluminum, carbon steel, copper alloy, and stainless steel cooling unit components that are exposed to a ventilation environment or to an uncontrolled raw water environment from cooling unit drain pans, and of certain heat exchanger components exposed to treated water or ventilation environments in the Control Structure Chilled Water, the Primary Containment Atmosphere Circulation, and the Control Structure and Reactor Building HVAC systems. The inspection provides direct evidence as to whether, and to what extent, a loss of material due to crevice, galvanic, general, or pitting corrosion, or reduction in heat transfer due to fouling of heat exchanger tubes and fins, has occurred or is likely to occur in these systems that could result in a loss of intended function.

The Cooling Units Inspection is a new one-time inspection that will be implemented prior to the period of extended operation. The inspection activities will be conducted within the 10-year period prior to the period of extended operation.

3.14.2.17 Crane Inspection Program (NUREG 1931, Section 3.0.3.1.8)

The Crane Inspection Program is an existing program that manages loss of material for cranes (including bridge, trolley, rails, and girders), monorails, and hoists within the scope of license renewal. The Crane Inspection Program is based on guidance contained in ANSI B30.2 for overhead and gantry cranes, ANSI B30.11 for monorail systems and underhung cranes, and ANSI B30.16 for overhead hoists. The inspections monitor structural members for signs of corrosion other than minor surface corrosion.

3.14.2.18 Fire Protection Program (NUREG 1931, Section 3.0.3.2.8)

The Fire Protection Program is an existing program that is described in the Fire Protection Review Report (FPRR) and which is credited with aging management of components with fire barrier functions in the scope of license renewal. Periodic visual inspections and functional tests are performed, as appropriate, of fire dampers, fire barrier walls, ceilings and floors, fire rated penetration seals (fire stops), fire wraps, fireproofing, and fire doors to ensure that functionality and operability are maintained. The Fire Protection Program is a condition monitoring program, comprised of tests and inspections generally in accordance with the applicable National Fire Protection Association (NFPA) recommendations.

3.14.2.19 Fire Water System Program (NUREG 1931, Section 3.0.3.2.9)

The Fire Water System Program (sub-program of the overall Fire Protection Program) is an existing program that is described in the Fire Protection Review Report (FPRR) and which is credited with aging management of the water suppression components in the scope of license renewal. Periodic inspection and testing of the water-based fire suppression systems provides reasonable assurance that the systems will remain capable of performing their intended function. Periodic inspection and testing activities include hydrant and hose station inspections, fire main flushing, flow tests, and sprinkler inspections. The Fire Water System Program is a condition monitoring program, comprised of tests and inspections generally in accordance with the applicable National Fire Protection Association (NFPA) recommendations.

Prior to the period of extended operation, the Fire Water System Program will be enhanced to incorporate:

a) Sprinkler head sampling/replacements, in accordance with NFPA 25;

b) Ultrasonic testing of representative above ground portions of water suppression piping that are exposed to water but which do not normally experience flow, are associated with a dry-pipe sprinkler system and may contain stagnant water, or is pre-action or deluge piping that is normally dry but may have been wetted and not completely dried;

c) At least one visual inspection (opportunistic or focused) of the internal surface of buried fire water piping within the 10 year period prior to the period of extended operation; and

d) At least one inspection per year of 'wet' fire protection piping for wall thickness and pipe blockage, if no opportunistic inspection is completed.

3.14.2.20 Flow-Accelerated Corrosion (FAC) Program (NUREG 1931, Section 3.0.3.1.7)

The Flow-Accelerated Corrosion (FAC) Program is an existing program that manages loss of material for carbon steel components located in systems that are susceptible to flow-accelerated corrosion, also called erosion/corrosion. The Flow-Accelerated Corrosion (FAC) Program is a condition monitoring program which ensures that the integrity of piping systems susceptible to flow-accelerated corrosion is maintained. The program was developed in response to NRC Bulletin 87-01 and NRC Generic Letter 89-08. The Flow-Accelerated Corrosion (FAC) Program follows the guidance and recommendations of EPRI NSAC-202L and combines the elements of predictive analysis, inspections (to baseline and monitor wall-thinning), industry experience, station information gathering and communication, and engineering judgment to monitor and predict flow-accelerated corrosion wear rates.

3.14.2.21 Fuel Oil Chemistry Program (NUREG 1931, Section 3.0.3.2.11)

The Fuel Oil Chemistry Program is an existing program that maintains fuel oil quality in order to mitigate damage due to loss of material and cracking of susceptible materials for plant components that are within the scope of license renewal and that contain fuel oil. The program manages the relevant conditions that could lead to the onset and propagation of loss of material or cracking through proper monitoring and control of fuel oil contamination consistent with pertinent plant technical specifications/requirements and American Society for Testing of Materials (ASTM) standards for fuel oil. The relevant conditions are specific contaminants such as water or microbiological organisms in the fuel oil that could lead to corrosion or stress corrosion cracking (SCC) of susceptible materials. Exposure to these contaminants is minimized by verifying the quality of new fuel oil before it enters the storage tanks and by periodic sampling to ensure that the tanks are free of water and particulates. The Fuel Oil Chemistry Program is a mitigation program.

The Fuel Oil Chemistry Program is supplemented by the Chemistry Program Effectiveness Inspection which provides verification of the effectiveness of the Fuel Oil Chemistry Program in mitigating the effects of aging.

3.14.2.22 Heat Exchanger Inspection (NUREG 1931, Section 3.0.3.1.12)

The Heat Exchanger Inspection detects and characterizes the condition of heat exchanger tubes in the Control Structure Chilled Water (CSCW), Fire Protection (FP) High Pressure Coolant Injection (HPCI), and Reactor Core Isolation Cooling (RCIC) systems.

The scope of the Heat Exchanger Inspection includes the CSCW chiller oil cooler and chiller evaporator, the FP diesel engine driven fire pump heat exchanger and lube oil cooler, and the HPCI and RCIC lube oil coolers. The inspection provides direct evidence as to whether, and to what extent, cracking due to SCC or reduction in heat transfer due to fouling has occurred or is likely to occur that could result in a loss of intended function.

The Heat Exchanger Inspection is a new one-time inspection that will be implemented prior to the period of extended operation. The inspection activities will be conducted within the 10-year period prior to the period of extended operation.

3.14.2.23 Inservice Inspection (ISI) Program (NUREG 1931, Section 3.0.3.2.1)

The Inservice Inspection (ISI) Program is an existing program that manages cracking and loss of material of multiple reactor coolant system pressure boundary components, including the reactor vessel, a limited number of internals components, and the reactor coolant system pressure boundary. The Inservice Inspection (ISI) Program was developed as required by 10 CFR 50.55a. The program is in accordance with the requirements detailed in the ASME Boiler and Pressure Vessel Code, Section XI, Division 1, Subsections IWA, IWB, IWC, IWD, IWE, IWF, IWL, Mandatory Appendices, and approved ASME Code Cases.

The inservice inspections conducted throughout the service life of SSES will comply, to the extent practical, with the requirements of the ASME Code Section XI Edition and Addenda incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the inspection interval, subject to prior approval via letter from the NRC. This is consistent with NRC statements of consideration (SOC) for 10 CFR 54 associated with the adoption of new editions and addenda of the ASME Code in 10 CFR 50.55a.

3.14.2.24 Inservice Inspection (ISI) Program – IWE (NUREG 1931, Section 3.0.3.1.19)

The Inservice Inspection (ISI) Program – IWE is an existing program that establishes responsibilities and requirements for conducting IWE inspections as required by 10 CFR 50.55a. The Inservice Inspection (ISI) Program – IWE includes visual examination of all accessible surface areas of the steel liner for the reinforced concrete Primary Containment and its integral attachments, and containment pressure-retaining bolting in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, Division 1 for Subsection IWE.

The inservice examinations conducted throughout the service life of SSES will comply, to the extent practical, with the requirements of the ASME Section XI Edition and Addenda incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the inspection interval, subject to prior approval via letter from the NRC. This is consistent with NRC statements of consideration (SOC), for 10 CFR 54, associated with the adoption of new editions and addenda of the ASME Code in 10 CFR 50.55a.

3.14.2.25 Inservice Inspection (ISI) Program – IWF (NUREG 1931, Section 3.0.3.1.21)

The Inservice Inspection (ISI) Program – IWF is an existing program that establishes responsibilities and requirements for conducting IWF Inspections for ASME Class 1, 2, and 3 component supports as required by 10 CFR 50.55a. The Inservice Inspection (ISI) Program – IWF visual examination for supports is based on sampling of the total support population. The sample size varies depending on the ASME Class. The largest sample size is specified for the most critical supports (ASME Class 1). The sample size decreases for the less critical supports (ASME Class 2 and 3). The primary inspection method employed is visual examination. Degradation that potentially compromises support function or load capacity is identified for evaluation. Supports requiring corrective actions are re-examined during the next inspection period in accordance with the requirements of ASME Section XI, Division 1 (edition and addenda described in 3.14.2.23) for Subsection IWF.

The inservice examinations conducted throughout the service life of SSES will comply, to the extent practical, with the requirements of ASME Section XI Edition and Addenda incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the inspection interval, subject to prior approval via letter from the NRC. This is consistent with NRC statements of consideration (SOC), for 10 CFR 54, associated with the adoption of new editions and addenda of the ASME Code in 10 CFR 50.55a.

3.14.2.26 Inservice Inspection (ISI) Program - IWL (NUREG 1931, Section 3.0.3.1.20)

The Inservice Inspection (ISI) Program – IWL is an existing program that establishes responsibilities and requirements for conducting IWL Inspections as required by 10 CFR 50.55a. The Inservice Inspection (ISI) Program – IWL includes visual examination of all accessible surface areas of the reinforced concrete Primary Containment in accordance with the requirements of ASME Section XI, Division 1 (edition and addenda described in 3.14.2.23) for Subsection IWL.

No applicable aging effects have been identified for the Primary Containment concrete. However, the Inservice Inspection (ISI) Program – IWL will be used to confirm the absence of significant aging effects for the extended period of operation.

The inservice examinations conducted throughout the service life of SSES will comply, to the extent practical, with the requirements of ASME Section XI Edition and Addenda incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the inspection interval, subject to prior approval via letter from the NRC. This is consistent with NRC statements of consideration, for 10 CFR 54, associated with the adoption of new editions and addenda of the ASME Code in 10 CFR 50.55a.

3.14.2.27 Leak Chase Channel Monitoring Activities (NUREG 1931, Section 3.0.3.3.2)

The Leak Chase Channel Monitoring Activities is an existing program that consists of observation and surveillance activities to detect leakage from the spent fuel pool and the fuel shipping cask storage pool liners due to aging and age-related degradation. The Leak Chase Channel Monitoring Activities is a condition monitoring program.

3.14.2.28 Lubricating Oil Analysis Program (NUREG 1931, Section 3.0.3.2.15)

The Lubricating Oil Analysis Program is an existing program that mitigates damage due to loss of material and reduction of heat transfer due to fouling for plant components that are within the scope of license renewal and exposed to lubricating oil. The program manages the relevant conditions that could lead to the onset and propagation of a loss of material, or reduction in heat transfer for heat exchanger tubes, through proper monitoring consistent with manufacturer's recommendations, the equipment's importance to safe plant operation, equipment accessibility and American Society for Testing of Materials (ASTM) standards for lubricating oil. The relevant conditions are specific parameters including particulate and water concentrations, viscosity, neutralization number, and flash point that could lead to, or are indicative of, conditions for age-related degradation of susceptible materials. The Lubricating Oil Analysis Program is a mitigation program.

Prior to the period of extended operation, the Lubricating Oil Analysis Program will be enhanced to include sampling of the lubricating oil from the Control Structure Chiller, Reactor Building Chiller, and Diesel Engine Driven Fire Pumps when the oil is changed. The oil will be tested for water and, excluding the Diesel Engine Dirven Fire Pumps, for particle count. The Diesel Engine Driven Fire Pumps will be tested via direct read ferrography and spectrochemical testing instead of particle counting due to the condition of the engine oil upon changing.

The Lubricating Oil Analysis Program is supplemented by the Lubricating Oil Inspection which provides verification of the effectiveness of the Lubricating Oil Analysis Program in mitigating the effects of aging.

3.14.2.29 Lubricating Oil Inspection (NUREG 1931, Section 3.0.3.1.13)

The Lubricating Oil Inspection detects and characterizes the condition of materials in systems and components for which the Lubricating Oil Analysis Program is credited with aging management. The inspection provides direct evidence as to whether, and to what extent, a loss of material or a reduction in heat transfer due to fouling has occurred.

The Lubricating Oil Inspection is a new one-time inspection that will be implemented prior to the period of extended operation. The inspection activities will be conducted within the 10-year period prior to the period of extended operation.

3.14.2.30 Not Used (NUREG 1931, Section 3.0.3.1.14)

3.14.2.31 Masonry Wall Program (NUREG 1931, Section 3.0.3.2.16)

The Masonry Wall Program is an existing program that consists of inspection activities to detect cracking of masonry walls within the scope of license renewal. Masonry walls that perform a fire barrier intended function are also managed by the Fire Protection Program. The Masonry Wall Program is implemented as part of the Structures Monitoring Program. The Masonry Wall Program performs visual inspection of external surfaces of masonry walls.

Prior to the period of extended operation, the Masonry Wall Program will be enhanced to specify that for each masonry wall, the extent of observed masonry cracking and/or degradation of steel edge supports/bracing be evaluated to ensure that the current evaluation basis is still valid.

3.14.2.32 Metal-Enclosed Bus Inspection Program (NUREG 1931, Section 3.0.3.1.26)

The Metal-Enclosed Bus Inspection Program manages the aging of the metal-enclosed bus within the scope of license renewal. The program provides for inspection of the applicable metal-enclosed bus on a 10-year interval, in order to determine if age-related degradation is occurring.

The Metal-Enclosed Bus Inspection Program is a new aging management program that will be implemented prior to the period of extended operation.

3.14.2.33 Monitoring and Collection System Inspection (NUREG 1931, Section 3.0.3.1.15)

The Monitoring and Collection System Inspection detects and characterizes the conditions on the internal surfaces of subject components that are exposed to equipment/area drainage water and other potential contaminants/fluids. The inspection provides direct evidence as to whether, and to what extent, a loss of material has occurred or is likely to occur in the Liquid Waste Management System that could result in a loss of intended function.

The Monitoring and Collection System Inspection is a new one-time inspection that will be implemented prior to the period of extended operation. The inspection activities will be conducted within the 10-year period prior to the period of extended operation.

3.14.2.34 Non-EQ Cables and Connections Used in Low-Current Instrumentation Circuits Program (NUREG 1931, Section 3.0.3.1.24)

The Non-EQ Cables and Connections Used in Low-Current Instrumentation Circuits Program manages the age-related degradation associated with non-EQ, low-current instrumentation cables and connections within the scope of license renewal.

The program applies to in-scope, non-EQ electrical cables and connections used in neutron monitoring and radiation monitoring circuits with sensitive, low-current signals. The Non-EQ Cables and Connections Used in Low-Current Instrumentation Circuits Program will perform testing of the applicable cable systems to identify reduction in insulation resistance. The tests will be performed at least every ten years, with the frequency to be determined by engineering evaluation.

The Non-EQ Cables and Connections Used in Low-Current Instrumentation Circuits Program is a new aging management program that will be implemented prior to the period of extended operation.

3.14.2.35 Non-EQ Electrical Cables and Connections Visual Inspection Program (NUREG 1931, Section 3.0.3.1.23)

The Non-EQ Electrical Cables and Connections Visual Inspection Program manages the aging of non-EQ electrical cables and connections within the scope of license renewal. The program provides for visual inspection on a 10-year interval of accessible, non-EQ electrical cables and connections, in order to determine if age-related degradation is occurring, particularly in plant areas with high temperatures and/or high radiation levels.

The Non-EQ Electrical Cables and Connections Visual Inspection Program is a new aging management program that will be implemented prior to the period of extended operation.

3.14.2.36 Non-EQ Inaccessible Medium-Voltage Cables Program (NUREG 1931, Section 3.0.3.1.25)

The Non-EQ Inaccessible Medium-Voltage Cables Program manages the aging of non-EQ inaccessible medium-voltage electrical cables subject to wetting within the scope of license renewal. The program provides for the periodic testing of non-EQ inaccessible medium-voltage electrical cables, in order to determine if age-related degradation is occurring, and includes provision for the inspection of associated manholes to identify any collection of water. The cable testing frequency will be based on plant operating experience, but will be performed at least once every ten years. The electrical manhole inspection frequency will be based on plant operating experience, but will be performed at least once every two years.

The Non-EQ Inaccessible Medium-Voltage Cables Program is a new aging management program that will be implemented prior to the period of extended operation.

3.14.2.37 Non-EQ Electrical Cable Connections Program (NUREG 1931, Section 3.0.3.1.27)

The Non-EQ Electrical Cable Connections Program manages the aging for the metallic parts of non-EQ electrical cable connections within the scope of license renewal. The program addresses cable connections that are used to connect cable conductors to other cables or electrical devices. Aging management for the metallic parts of the non-EQ electrical cable connections that are subject to aging stressors will be provided by testing. A representative sample of non-EQ electrical cable connections will be selected for testing, considering the effects of their application (high, medium, and low voltage), circuit loading, and location with respect to electrical connection stressors. Thermography will be used to test a representative sample of cable connections to provide an indication of the integrity of the connections. The tests will be performed at least every ten years, with the frequency to be determined by engineering evaluation.

The Non-EQ Electrical Cable Connections Program is a new aging management program that will be implemented prior to the period of extended operation.

3.14.2.38 Piping Corrosion Program (NUREG 1931, Section 3.0.3.2.6)

The Piping Corrosion Program is an existing program that manages fouling due to particulates (e.g., corrosion products) and biological material (micro- and/or macro-organisms), and loss of material due to corrosion and erosion for components located in systems within the scope of the program that are exposed to a raw water environment. The program also manages the applicable aging effects for the internal environments of heat exchanger components within the scope of the program.

The Piping Corrosion Program is a combination of condition monitoring program (consisting of inspections, surveillances, and testing to detect the presence of, and to assess the extent of, damaged coatings, fouling and loss of material) and a mitigation program (consisting of chemical treatments and cleaning activities to minimize fouling and loss of material, and use of protective coatings in areas vulnerable to erosion).

Prior to the period of extended operation, the Piping Corrosion Program will be enhanced to include the Standby Gas Treatment System loop seals, and to also incorporate performance, documentation and trending of opportunistic visual inspections (during normal maintenance/repair activities).

3.14.2.39 Preventive Maintenance Activities – RCIC/HPCI Turbine Casings (NUREG 1931, Section 3.0.3.3.3)

The Preventive Maintenance Activities – RCIC/HPCI Turbine Casings is an existing program that manages loss of material on the internal surfaces of the Reactor Core Isolation Cooling (RCIC) and High Pressure Coolant Injection (HPCI) pump turbine casings, and on the internal surfaces of associated piping and piping components (rupture disks and valve bodies), that are constructed of carbon steel or cast iron.

The Preventive Maintenance Activities – RCIC/HPCI Turbine Casings is a condition monitoring program, consisting of inspections and surveillance activities to detect aging and age-related degradation.

Prior to the period of extended operation, the Preventive Maintenance Activities – RCIC / HPCI Turbine Casings will be enhanced to incorporate:

a) A specific step to perform a visual inspection of the RCIC turbine casing.

b) Performance of inspections by qualified personnel using VT-3 or equivalent inspection methods, and reporting and trending of inspection results.

c) Specific acceptance criteria for inspections.

3.14.2.40 Reactor Head Closure Studs Program (NUREG 1931, Section 3.0.3.1.2)

The Reactor Head Closure Studs Program is an existing program that manages cracking for the reactor head closure studs. The Reactor Head Closure Studs Program includes (a) inservice inspection in conformance with the requirements of ASME Code, Section XI, Subsection IWB (edition and addenda described in 3.14.2.23), Table IWB 2500-1, and (b) preventive measures in accordance with Regulatory Guide 1.65 to mitigate cracking. The Reactor Head Closure Studs Program is implemented by the design of the plant and the Inservice Inspection (ISI) Program.

3.14.2.41 Reactor Vessel Surveillance Program (NUREG 1931, Section 3.0.3.2.12)

The Reactor Vessel Surveillance Program is an existing program that manages reduction of fracture toughness for the low alloy steel reactor vessel shell and welds in the beltline region. The Reactor Vessel Surveillance Program is a condition monitoring program developed in response to 10 CFR 50 Appendix H.

The SSES Reactor Vessel Surveillance Program is part of the Integrated Surveillance Program (ISP) described in BWRVIP-78, BWRVIP-86-A and BWRVIP-116, and approved by the NRC staff. BWRVIP-116 extends the ISP to cover the period of extended operation. SSES will follow the requirements of the BWRVIP ISP and will apply the ISP data to SSES Units 1 and 2. The NRC approved the use of the BWRVIP ISP in place of a unique plant program at SSES.

The SSES Reactor Vessel Surveillance Program will be enhanced, as necessary, to ensure that additional requirements that are specified in the NRC safety evaluation dated March 1, 2006, for BWRVIP-116 will be addressed before the period of extended operation. The program will include a requirement that, if a standby capsule is removed from either of the SSES Unit 1 or Unit 2 reactor vessels, without the intent to test it, the capsule will be stored in a manner which maintains it in a condition which would permit its future use, including during the period of extended operation, if necessary,

3.14.2.42 RG 1.127 Water-Control Structures Inspection (NUREG 1931, Section 3.0.3.2.18)

The RG 1.127 Water-Control Structures Inspection is an existing program that consists of inspection and surveillance activities to detect aging and age-related degradation. The RG 1.127 Water-Control Structures Inspection ensures the structural integrity and operational adequacy of the Spray Pond (including concrete liner, emergency spillway, and riser encasements), ESSW Pumphouse (including pump intake chambers, overflow weir and chamber, and structural components within the ESSW Pumphouse), and the earthen embankments along the Spray Pond.

Prior to the period of extended operation, the RG 1.127 Water-Control Structures Inspection will be enhanced to add the Spray Pond (including concrete liner, emergency spillway, riser encasements, and earthen embankments) to its scope for inspection. The program will be enhanced to include RG 1.127 inspection elements and degradation mechanisms for water-control structure inspection and to include acceptance criteria for water-control structures.

3.14.2.43 Selective Leaching Inspection (NUREG 1931, Section 3.0.3.1.17)

The Selective Leaching Inspection detects and characterizes the conditions on internal and external surfaces of subject components. The inspection provides direct evidence through a combination of visual examination and hardness testing of whether, and to what extent, a loss of material due to selective leaching has occurred or is likely to occur that could result in a loss of intended function.

The Selective Leaching Inspection is a new one-time inspection that will be implemented prior to the period of extended operation. The inspection activities will be conducted within the 10-year period prior to the period of extended operation.

3.14.2.44 Small Bore Class 1 Piping Inspection (NUREG 1931, Section 3.0.3.1.18)

The Small Bore Class 1 Piping Inspection is a one-time inspection to detect cracking resulting from thermal and mechanical loading or intergranular stress corrosion. The inspection will provide assurance that cracking of small bore Class 1 piping is not occurring or an evaluation of any detected crack indications will be performed to justify continued operation with no further monitoring, such that an aging management program (AMP) is not warranted. The inspection will also confirm the effectiveness of the BWR Water Chemistry Program mitigating cracking due to intergranular stress corrosion. The Small Bore Class 1 Piping Inspection is applicable to small bore ASME Code Class 1 piping and systems less than 4 inches nominal pipe size (NPS 4), which includes pipes, fittings, and branch connections.

The Small Bore Class 1 Piping Inspection is a new one-time inspection that will be implemented prior to the period of extended operation. The inspection activities will be conducted within the 10-year period prior to the period of extended operation.

3.14.2.45 Structures Monitoring Program (NUREG 1931, Section 3.0.3.2.17)

The Structures Monitoring Program is an existing program that manages age-related degradation of plant structures and structural components within its scope to ensure that each structure or structural component retains the ability to perform its intended function. Aging effects are detected by visual inspection of external surfaces prior to the loss of the structure's or component's intended function.

This program implements provisions of the Maintenance Rule, 10 CFR 50.65, that relate to structures, masonry walls, and water-control structures. Concrete and masonry walls that perform a fire barrier intended function are also managed by the Fire Protection Program.

Prior to the period of extended operation, the Structures Monitoring Program will be enhanced to include structures within the scope of license renewal. The program will be enhanced to specify inspections of a below grade structural wall or structural component that becomes accessible through excavation. The program will be enhanced to clarify the component types included as “structural components”. The program will be enhanced to specify degradation mechanisms for elastomer and earthen embankment inspection. The program will be enhanced to include RG 1.127 inspection elements for water-control structures. The program will be enhanced to include requirements for review of site groundwater and raw water parameters. The program will also be enhanced to specify inspection requirements for masonry walls. The program will be enhanced to include direction for quantifying, monitoring and trending of inspection results, guidance for inspection reporting, data collection and documentation, acceptance criteria and critical parameters to monitor degradation and to trigger level of inspection and initiation of corrective action; and better alignment with referenced Industry codes, standards and guidelines. The program will be enhanced to include specific qualification requirements for the inspector.

3.14.2.46 Supplemental Piping/Tank Inspection (NUREG 1931, Section 3.0.3.1.16)

The Supplemental Piping/Tank Inspection detects and characterizes the condition of carbon steel, stainless steel and cast iron components that are exposed to moist air environments, particularly the aggressive alternate wet/dry environment that exists at air-water interfaces, and for internal surfaces of diesel exhaust components due to periodic exposure to exhaust gases containing moisture and contaminants. The inspection provides direct evidence as to whether, and to what extent, a loss of material or cracking (of stainless steel exposed to diesel exhaust) has occurred or is likely to occur that could result in a loss of intended function.

The Supplemental Piping/Tank Inspection is a new one-time inspection that will be implemented prior to the period of extended operation. The inspection activities will be conducted within the 10-year period prior to the period of extended operation.

3.14.2.47 System Walkdown Program (NUREG 1931, Section 3.0.3.2.14)

The System Walkdown Program is an existing program that manages the following aging effects for the external surfaces, and in some cases the internal surfaces, of mechanical components within the scope of license renewal:

a) Loss of material for metals that are exposed to indoor air, outdoor air, or ventilation environments, including both the HVAC-type internal environments and ambient air internal environments, such as that found in the upper portion of a vented tank.

b) Cracking and/or change in material properties for elastomers (neoprene and rubber) and polymers (Teflon) that are exposed to indoor air or ventilation environments.

The System Walkdown Program is a condition monitoring program, consisting of observation and surveillance activities to detect aging and age-related degradation.

Prior to the period of extended operation, the System Walkdown Program will be enhanced to include the license renewal systems that contain mechanical components whose external surfaces require aging management during the period of extended operation. The program will also be enhanced to address opportunistic inspections of normally inaccessible components (e.g., those that are insulated), and those that are accessible only during refueling outages. The program will also be enhanced by addition of a routine activity to inspect elastomers and polymers for cracking and/or change in material properties and to include inspection of other metals, copper alloy and stainless steel. The program will also be enhanced to sample normally inaccessible components in underground vaults, pits, and manholes. In addition, the program will be enhanced to include a visual and ultrasonic inspection of the external surfaces of piping passing into structures through penetrations (underground piping) for those penetrations with a history of leakage.

3.14.2.48 Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steel (CASS) Program (NUREG 1931, Section 3.0.3.1.6)

The Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steel (CASS) Program augments the visual inspection of the reactor vessel internals done in accordance with the ASME Code, Section XI, Subsection IWB, Category B-N-2. The inspection is augmented to detect the effects of loss of fracture toughness due to thermal aging and neutron irradiation embrittlement of cast austenitic stainless steel (CASS) reactor vessel internal components. The aging management program includes (a) identification of susceptible components determined to be limiting from the standpoint of thermal aging susceptibility (i.e., ferrite and molybdenum contents, casting process, and operating temperature) and/or neutron irradiation embrittlement (neutron fluence), and (b) for each potentially susceptible component, aging management is accomplished through either a supplemental examination of the affected component based on the neutron fluence to which the component has been exposed as part of the SSES 10-year Inservice Inspection (ISI) Program during the license renewal term, or a component-specific evaluation to determine its susceptibility to loss of fracture toughness.

The Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steel (CASS) Program is a new aging management program that will be implemented prior to the period of extended operation.

3.14.2.49 Fatigue Monitoring Program (NUREG 1931, Section 3.0.3.2.19)

The Fatigue Monitoring Program manages fatigue for all Class 1 components, including the reactor pressure vessel. In order not to exceed the design basis limit on fatigue usage, the aging management program monitors and tracks the number and severity of critical thermal and pressure transients and calculates the fatigue usage for the limiting locations of the reactor coolant pressure boundary.

Prior to the period of extended operation the program will be enhanced by adding the following required actions:

a) The program will verify that components which have satisfied ASME Section III, Paragraph N-415.1 requirements (i.e., RPV nozzles N6A, N6B, and N7) continue to satisfy these requirements prior to and during the period of extended operation, thereby allowing fatigue to be continued to be addressed under N-415.1.

b) The program will review Class 1 valve fatigue analyses and other fatigue-related TLAA, such as flued head analyses and high energy line break evaluations, when sufficient fatigue accumulation has occurred, to determine if additional actions are required to address fatigue-related concerns.

c) The program will define specific fatigue usage values for all monitored locations, including those locations that account for the effect of the reactor water environment that, if reached, will require further action. These fatigue usage values shall be conservatively set to values that will allow for not less than 4 years of additional plant operation before the actual fatigue usage at any location would reach the design basis limit. Upon reaching the defined usage at a location, the program will require an action request to be generated. The action request will require further engineering evaluation to resolve the issue.

d) The program will implement one or more of the following actions, if fatigue usage at a monitored location, including any location that accounts for the effect of the reactor water environment, is projected to reach the design basis limit prior to the end of the period of extended operation:

    1) Further refinement of the fatigue analyses to lower the CUFs to less than the allowable;

    2) Repair of the affected components;

    3) Replacement of the affected components;

    4) Management by an inspection program that has been reviewed and approved by the NRC.

3.14.2.50 Preventive Maintenance Activities – Main Turbine Casing (NUREG 1931, Section 3.0.3.3.4)

The Preventive Maintenance Activities – Main Turbine Casing is an existing program that manages loss of material due to flow-accelerated corrosion on the internal surfaces of the high pressure casing for the main turbine.

The Preventive Maintenance Activities – Main Turbine Casing is a condition monitoring program consisting of inspections performed to detect aging and age-related degradation.

Prior to the period of extended operation, the Preventative Maintenance Activities – Main Turbine Casing will be enhanced to specify that the inspection of the high pressure turbine shell will consist of a VT-3 or equivalent visual inspection of accessible surfaces and an ultrasonic examination of selected locations for wall thickness.

3.14.2.51 Fuse Holders Program (NUREG 1931, Section 3.0.3.2.20)

The Fuse Holders Program is a new aging management program that manages increased connection resistance due to fatigue of fuse holder clamps. The program provides for periodic inspection of fuse holder clamps within the scope of license renewal that are not in enclosures containing active components and whose fuses are scheduled for removal once every 12 months, or more frequently.

The Fuse Holders Program is a condition monitoring program consisting of inspections performed on a 10-year frequency to detect aging and age-related degradation.

The Fuse Holders Program is a new aging management program that will be implemented prior to the period of extended operation.

3.14.3 EVALUATION OF TIME-LIMITED AGING ANALYSES (TLAA)

In accordance with 10 CFR 54.21(c), an application for a renewed operating license requires an evaluation of time-limited aging analyses (TLAA) for the period of extended operation. The license renewal evaluation of time-limited aging analyses (TLAA) identified existing aging management programs necessary to provide reasonable assurance that 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. This section identifies the TLAA, summarizes the evaluation of each, and, where necessary, describes the aging management programs that will be required. These aging management programs will be implemented during the period of extended operation. One-time inspections will be conducted within the 10-year period prior to beginning the period of extended operation.

Three elements of an effective aging management program that are common to all aging management programs are corrective actions, confirmation process, and administrative controls. These elements are included in the SSES Operational Quality Assurance (OQA) Program, which implements the requirements of 10 CFR 50, Appendix B. Prior to the period of extended operation, the elements of corrective action, confirmation process, and administrative controls in the SSES OQA Program will be applied to required aging management programs for both safety-related and nonsafety-related structures and components determined to require aging management during the period of extended operation.

3.14.3.1 Reactor Vessel Neutron Embrittlement

The ferritic materials of the reactor vessel are subject to embrittlement due to high energy neutron exposure. Reactor vessel neutron embrittlement is a TLAA. Neutron fluence, upper shelf energy, adjusted reference temperature, and vessel pressure-temperature limits are time-dependent items that must be investigated to evaluate vessel embrittlement.

3.14.3.1.1 Neutron Fluence

High energy (>1 MeV) 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 SSES Units 1 and 2. Use of this methodology for evaluations of fluence for the SSES units 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 54 effective full power years (EFPY), i.e., at the end of 60 years of operation. Using actual reactor core power histories to-date and conservative estimates of future core designs for each unit, extended operation to 60 years will be bounded by 54 EFPY.

Neutron fluence is not a TLAA, it is a time-limited assumption used in various neutron embrittlement TLAA.

3.14.3.1.2 Upper Shelf Energy Evaluation

10 CFR 50, Appendix G requires that upper shelf energy (USE) values for reactor pressure vessel materials include the effects of neutron radiation. It states that USE for the beltline materials including plates and welds be maintained at no less than 50 ft-lb for the life of the reactor vessel. Calculated fluence values for extended power uprate (EPU) and extended operation to 54 EFPY exceed previously determined fluence values based on materials surveillance program information for Units 1 and 2. Therefore, projections of changes in USE for the period of extended operation are required in accordance with 10 CFR 50, Appendix G.

The projections of changes in USE for the period of extended operation for the RPV beltline plates and welds for Units 1 and 2 were determined in accordance with Regulatory Guide (RG) 1.99, Revision 2. For the plates and welds with projected USE values of 50 ft-lb or greater at 54 EFPY, the criterion of 10 CFR 50, Appendix G, has been met and no further evaluation is required.

For plates and welds that do not meet the 50 ft-lb criterion, the equivalent margin analyses (EMA) documented in BWRVIP-74-A were used to demonstrate that the 54 EFPY USE values remain in compliance with 10 CFR 50, Appendix G. As prescribed in BWRVIP-74-A, the predicted decrease in USE from RG 1.99, Figure 2 was compared to the decrease assumed in the EMA for each vessel beltline plate and weld that fails to meet the 50 ft-lb criterion. The results demonstrate that all evaluated plates and welds are bounded by the BWRVIP-74-A equivalent margin analyses.

Therefore, the effects of neutron radiation have been evaluated, and all RPV beltline materials for Units 1 and 2 have been demonstrated to remain in compliance with Appendix G of 10 CFR 50 for the period of extended operation.

3.14.3.1.3 Adjusted Reference Temperature (ART) Analysis

In addition to USE, the other key parameter that characterizes the fracture toughness of a material is the reference temperature for nil-ductility transition (RT_{NDT}). This reference temperature will change as its exposure to neutron radiation increases. The effects of neutron fluence on RT_{NDT} are reflected in the change in this reference temperature, ΔRT_{NDT}, and the resulting adjusted reference temperature, ART, is calculated by adding ΔRT_{NDT} to RT_{NDT} along with appropriate margin to account for uncertainties.

The methodology used to calculate ART for the vessel beltline plates and welds is provided in Regulatory Guide 1.99. The ART values projected to 54 EFPY are used to develop Pressure-Temperature (P-T) limit curves. There are no limits or specific acceptance criteria for the projected ART values.

3.14.3.1.4 Pressure-Temperature (P-T) Limits

To assure that adequate margins of safety are maintained for various modes of reactor operation, 10 CFR 50, Appendix G specifies pressure and temperature requirements for affected materials for the service life of the reactor vessel. The basis for these fracture toughness requirements is ASME Section XI, Appendix G. The ASME Code requires P-T limits be established for hydrostatic pressure tests and leak tests, for operation with the core not critical during heatup and cooldown, and for core critical operation.

Calculations were performed to develop P-T limit curves for SSES Units 1 and 2 for the period of extended operation (54 EFPY). The calculations were performed for the bounding regions of the reactor vessel to account for 54 EFPY fluence projections, which include the effects of EPU conditions. The P-T curves were developed in accordance with 10 CFR 50, Appendix G and the methodology in ASME Section XI, Appendix G. SSES will submit future P-T curve data to the NRC as necessary to comply with 10 CFR 50 Appendix G,

3.14.3.1.5 Reactor Vessel Circumferential Weld Examination Relief

BWRVIP-74-A reiterated the recommendation of BWRVIP-05 that reactor pressure vessel circumferential welds could be exempted from examination. The NRC safety evaluation report for BWRVIP-74 agreed, but required that plants apply for this relief request individually. The relief request should demonstrate that at the expiration of the current license, the circumferential welds satisfy the limiting conditional failure probability for circumferential welds in the (BWRVIP-05) evaluation. This evaluation of circumferential weld parameters is a TLAA.

SSES requested and received relief from circumferential vessel shell weld volumetric examinations. The SSES submittal included an analysis that showed that the reactor vessel parameters at 32 EFPY were within the bounding parameters for Chicago Bridge & Iron (CBI) vessels from the BWRVIP-05 safety evaluation report. As such, there is a lower conditional probability of failure for circumferential welds at SSES than that stated in the NRC's Final Safety Evaluation Report of BWRVIP-05.

The SSES reactor pressure vessel circumferential weld parameters at 54 EFPY will remain within the bounding parameters for CBI vessels at 64 EFPY from the BWRVIP-05 safety evaluation report. As such, the conditional probability of failure for circumferential welds remains below that stated in the safety evaluation report for BWRVIP-05.

SSES will process a relief request for circumferential vessel shell weld volumetric examinations for the period of extended operation in the same manner that has been the practice during the original licensing period.

3.14.3.1.6 Reactor Vessel Axial Weld Failure Probability

The NRC safety evaluation report for BWRVIP-74-A evaluated the failure frequency of axially oriented welds in BWR reactor pressure vessels, and determined failure frequency acceptance criteria for 40 years of reactor operation. Applicants for license renewal must evaluate axially oriented RPV welds to show that their failure frequency remains below the acceptance criteria calculated in the safety evaluation report for BWRVIP-74. An acceptable way to do this is to show that the mean RT_{NDT} of the limiting axial beltline weld at the end of the period of extended operation is less than the values specified in the safety evaluation report for BWRVIP-74.

The SSES axial weld mean RT_{NDT} at 54 EFPY is projected to be well below that in the SER, and thus the SSES axial weld failure frequency is well within the acceptable criteria.

3.14.3.1.7 Reflood Thermal Shock Analysis

FSAR Section 3.13.1, Regulatory Guide 1.2 “Thermal Shock to Reactor Pressure Vessels” addresses the possibility of brittle fracture of the reactor vessel resulting from reflooding of the vessel following a postulated loss of coolant accident. This concern is addressed in NEDO-10029, “An Analytical Study on Brittle Fracture of GE-BWR Vessels Subject to the Design Basis Accident,” in which a conservative analysis is documented. That document provides an upper bound limit on brittle fracture failure for the materials and concludes that catastrophic failure is not possible. However, the analysis performed in NEDO-10029 is only valid for 40 years of operation.

A more recent analysis provides a technical basis for addressing brittle fracture of BWR vessels due to vessel reflood following a design basis Loss of Coolant Accident (LOCA) during the period of extended operation. The analysis is documented in a paper by S. Ranganath of General Electric, entitled “Fracture Mechanics Evaluation of a Boiling Water Reactor Vessel Following a Postulated Loss of Coolant Accident,” which was presented as Paper G1/5 at the Fifth International Conference on Structural Mechanics in Reactor Technology in Berlin, Germany, in August 1979. While the analysis was performed for BWR-6 vessels, it is applicable to the SSES (BWR-4) vessels, based on the following:

a) It evaluated the main steam line break LOCA event, which is bounding for the evaluation of thermal stresses and brittle fracture in the vessel beltline region.

b) It applied to 251-inch, 238-inch, and 201-inch diameter BWR-6 vessels, since the structural details and operating conditions are similar. The SSES vessels are 251-inch diameter and the SSES response to a main steam line break LOCA is equivalent to that of the BWR-6 vessel (i.e., rapid depressurization and blowdown, immediately followed by ECCS injections to reflood the vessel).

c) It analyzed a 6-inch thick wall of a BWR-6 vessel. The SSES vessels are 6.1875 inches thick. A critical parameter of the fracture mechanics analysis is the wall temperature at a depth of 1/4 of the total thickness, 1/4T. The 1/4T location of the thinner BWR-6 vessel will cool faster than the 1/4T location of the SSES vessels. Thus, the BWR-6 analysis is conservative for the SSES vessels, since a lower temperature at the 1/4T location is worse for brittle fracture concerns.

The analysis presented in the Ranganath paper assumes end-of-life material toughness, which in turn depends on end-of-life adjusted reference temperature (ART). The analysis determined that the peak stress intensity for the LOCA event at the 1/4T location on the BWR-6 vessel would be 100 ksi√inches and that the available fracture toughness at the 1/4T location that coincides with the peak stress intensity would be a minimum of 200 ksi√inches. Thus, since the available toughness exceeds the applied stress intensity, an existing 1/4T flaw in the vessel wall would not propagate following a LOCA.

The BWR-6 analysis conclusion applies to the SSES vessels because the end-of-life (54 EFPY) ART values for the SSES vessels are such that the available material toughness at the 1/4T location of the SSES vessels would remain on the upper shelf of 200 ksi√inches, which exceeds the peak stress intensity of 100 ksi√inches for the analyzed event. Therefore, brittle fracture of the SSES vessels due to reflood thermal shock following a design basis LOCA is not possible during the period of extended operation.

3.14.3.2 Metal Fatigue

Fatigue evaluations for mechanical components are identified as TLAA; therefore, the effects of fatigue must be addressed for license renewal.

SSES monitors fatigue of the ASME Class 1 reactor coolant pressure boundary via the Fatigue Monitoring Program, which uses a computer program, FatiguePro, to count transient cycles and calculate fatigue usage.

Calculation of fatigue usage values is not required for non-Class 1 SSCs. Instead, stress intensification factors and lower stress allowables are used to ensure components are adequately designed for fatigue.

Certain components enveloped by the Primary Containment are also required to be evaluated for fatigue. These include penetrations, hatches, the drywell head, downcomer vents, safety relief valve (SRV) discharge piping, and SRV quenchers.

3.14.3.2.1 Reactor Pressure Vessel Fatigue Analysis

The design transients for the reactor pressure vessel (RPV) assembly are reported in FSAR Table 3.9-1. Design cumulative usage factors (CUFs) for the limiting reactor pressure vessel assembly locations are obtained from applicable design reports. These CUFs were calculated based on applicable design transients.

Metal fatigue for all reactor pressure vessel assembly components is managed by the existing SSES Fatigue Monitoring Program. This program includes requirements for continued monitoring and periodic updates to current and projected CUFs for the limiting reactor pressure vessel locations. The program will be enhanced to include an approach to address CUFs that will exceed the allowable before the end of the period of extended operation. The aging management approach will include one or more of the following, which is similar to the approach documented in ASME Code Section III Non-mandatory Appendix L:

a) Further refinement of the fatigue analyses to lower the CUFs to less than the allowable

b) Repair of the affected components

c) Replacement of the affected components

d) Management by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at intervals determined by a method accepted by the NRC)

The original RPV design report was not required to provide an explicit fatigue analysis for nozzles N6A, N6B, and N7, since the nozzles satisfied all requirements of ASME Section III, Paragraph N-415.1. As such, design CUFs were not calculated for these nozzles. The SSES Fatigue Monitoring Program will be enhanced to include a requirement to periodically determine if the requirements of N-415.1 remain satisfied, such that fatigue evaluations are not required for these nozzles prior to entering and during the period of extended operation.

The Fatigue Monitoring Program is credited for managing the effects of fatigue during the period of extended operation. Therefore, the TLAA associated with reactor pressure vessel fatigue are dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

3.14.3.2.2 Reactor Vessel Internals Fatigue Analyses

The Reactor Internals and Core Support Structures at SSES were designed in accordance with ASME Section III, Subsection NG. The fatigue evaluations performed to demonstrate the design adequacy of the internals for 40 years are TLAA.

Most recently, structural evaluations were performed to address the effects of operation under extended power uprate conditions and the extended period of plant operation to 60 years. The evaluations determined that the fatigue usage factors for all reactor pressure vessel internals remain within the ASME Section III Subsection NG allowable limits.

SSES also monitors the design transients using FatiguePro, as described above under Reactor Pressure Vessel Fatigue Analysis. This monitoring allows SSES to continually assess the potential for plant operating anomalies that could impact the assumptions made in the fatigue evaluations of plant components. In addition to plant transient monitoring, SSES has effectively implemented the inspection requirements of the BWRVIP program at SSES, as described in Section 3.14.2.10 above. These inspections provide further assurance that the effects of aging due to fatigue of the RPV internals will be managed during the period of extended operation.

Structural evaluations have demonstrated that fatigue usage will remain within design limits to the end of the period of extended operation. Also, the BWR Vessel Internals Program is credited for managing the aging effects of the reactor vessel internals during the period of extended operation. Therefore, the TLAA associated with fatigue of the reactor vessel internals are dispositioned in accordance with 10CFR54.21 (c)(1)(ii) and 10CFR54.21(c)(1)(iii).

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

Applicants for license renewal are required to address the reactor coolant environmental effects on fatigue of plant components. The minimum set of components is suggested to be the six (6) components defined in NUREG/CR-6260, as follows:

a) Reactor vessel shell and lower head

b) Reactor vessel feedwater nozzle

c) Reactor recirculation piping (including inlet and outlet nozzles)

d) Core spray line reactor vessel nozzle and associated Class 1 piping

e) Residual heat removal return line Class 1 piping

f) Feedwater line Class 1 piping

Calculation of a fatigue life adjustment factor, F_{en}, is determined for each fatigue-sensitive component. The environmental fatigue life adjustment factors are applied to the appropriate component CUFs to verify acceptability of the components for the period of extended operation.

Using fatigue data projected by the SSES Fatigue Monitoring Program and methodology accepted by the NRC, as noted above, SSES evaluated the limiting locations (a total of eleven component locations corresponding to the six NUREG/CR-6260 components), as appropriate for the material for each component location. Seven of the eleven locations evaluated have an environmentally adjusted CUF of greater than 1.0.

Prior to entering the period of extended operation, for each location that may exceed a CUF of 1.0 when considering environmental effects, SSES will implement one or more of the following:

a) Further refinement of the fatigue analyses to lower the CUFs to less than the allowable

b) Repair of the affected components

c) Replacement of the affected components

d) Management by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at intervals determined by a method accepted by the NRC)

Should SSES select the option to manage environmentally-assisted fatigue during the period of extended operation, details of the aging management program such as scope, qualification, method, and frequency will be provided to the NRC prior to the period of extended operation.

The effects of environmentally-assisted fatigue for the limiting locations identified in NUREG/CR-6260 have been evaluated. The effects of environmentally-assisted fatigue for these locations is addressed using one of the four approaches identified above.

The Fatigue Monitoring Program is credited for managing the effects of the reactor coolant environmental effects on fatigue during the period of extended operation. Therefore, the TLAA associated with environmentally-assisted fatigue has been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

3.14.3.2.4 Reactor Coolant Pressure Boundary Piping and Component Fatigue Analyses

The Class 1 boundary encompasses all reactor coolant pressure boundary piping (pipe and fittings) and in-line components subject to ASME Section XI, Subsection IWB, inspection requirements.

FSAR Section 3.9 provides details on the design transients to be considered in the fatigue analyses of reactor coolant pressure boundary (RCPB) components.

The SSES Fatigue Monitoring Program tracks the fatigue usage at the limiting locations throughout the RCPB. The use of FatiguePro and the SSES Fatigue Monitoring Program ensure that the fatigue of RCPB components is maintained below the ASME Code design limits.

All Class 1 valves are required to have a fatigue analysis. A review of a representative sample of Class 1 valve stress reports found the fatigue analyses to be conservatively simplistic, and the predicted fatigue was extremely low (less than 0.1). The simplified analyses for the valves do not provide the detailed information required to track fatigue usage by cycle counting or similar means. As an alternative, since the fatigue usage is typically much higher on the associated piping systems, and fatigue monitoring is performed for the limiting piping locations, the fatigue usage on the Class 1 valves is assumed to be bounded by the Class 1 piping locations. The fatigue on the valves will be managed indirectly by monitoring fatigue on the piping. If a piping system accumulates sufficient fatigue usage to indicate that design values are being approached, the Fatigue Monitoring Program will require a review of the valve fatigue analyses and other fatigue-related TLAA (such as flued head analyses and high energy line break evaluations) to determine if additional actions are required to address any of these additional fatigue-related concerns on the affected piping system.

Metal fatigue for all Class 1 reactor coolant pressure boundary piping and in-line components is managed by the SSES Fatigue Monitoring Program. This program includes requirements for continued monitoring and periodic updates to current and projected CUFs for the limiting piping locations. The program will be enhanced to include an approach to address CUFs that will exceed the allowable before the end of the period of extended operation. The aging management approach will include one or more of the following, which is similar to the approach documented in ASME Code Section III Non-mandatory Appendix L:

a) Further refinement of the fatigue analyses to lower the CUFs to less than the allowable

b) Repair of the affected components

c) Replacement of the affected components

d) Management by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at intervals determined by a method accepted by the NRC)

The Fatigue Monitoring Program is credited for managing the effects of fatigue during the period of extended operation. Therefore, the TLAA associated with fatigue of the reactor coolant pressure boundary piping and components have been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

3.14.3.3 Non-Class 1 Component Fatigue Analyses

Calculation of cumulative fatigue usage, i.e., CUFs, is not required for non-Class 1 components designed in compliance with the codes and standards 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 7000 full temperature cycles with reductions in allowable stresses as cycles increase beyond 7000. In addition, temperature thresholds above which fatigue should be considered for carbon steel and austenitic stainless steel are established.

The fatigue evaluation of non-Class 1 components determined whether the associated operating temperature exceeded threshold values for the affected materials and, if so, evaluated the number of transient cycles expected. In every case, the number of projected cycles for 60 years was found to be less than 7000 for piping and in-line components whose temperatures exceed threshold values. Therefore, fatigue for non-Class 1 piping and in-line components remains valid for the period of extended operation.

None of the non-Class 1 vessels, heat exchangers, storage tanks, or pumps were designed to ASME Section VIII, Division 2 or ASME Section III, Subsection NC-3200. Therefore, there is no fatigue TLAA for these components.

The TLAA associated with the fatigue of non-Class 1 components have been dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

3.14.3.4 Environmental Qualification of Electric Equipment (NUREG 1931, Section 3.0.3.1.28)

Environmental Qualification analyses for those components with a qualified life of 40 years or greater are identified as TLAA for SSES. NRC regulation 10 CFR 50.49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants” requires licensees to identify electrical equipment covered under this regulation and to maintain a qualification file demonstrating that the equipment is qualified for its application and will perform its safety function up to the end of its qualified life.

10 CFR 50.49 requires EQ components that are not qualified for the current license term to be refurbished, replaced, or have their qualifications extended prior to reaching the aging limits established in the aging evaluation. Reanalysis of aging evaluations to extend the qualifications of components is performed on a routine basis as part of the EQ Program. Important attributes for the reanalysis of aging evaluations include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, corrective actions (if acceptance criteria are not met), and the time remaining to the end of qualified life.

The SSES EQ Program is an existing program that implements the requirements of 10 CFR 50.49 and will be used to manage the effects of aging on the intended function(s) of the components associated with EQ TLAA for the period of extended operation.

3.14.3.5 Fatigue of Primary Containment, Attached Piping, and Components

3.14.3.5.1 ASME Class MC Components

FSAR Section 3.8.2.3.2.4 states the design thermal cycles for containment ASME Class MC stainless steel components, which includes the containment penetrations, hatches, and drywell head, to be 500 cycles for plant startup and shutdown and one cycle for a design basis accident. The reactor pressure vessel assembly and internal components are designed for 117 startups and 111 shutdowns for a combined total of 228 events. The maximum projected cycles for extended life to 60 years includes 148 startups and 148 shutdowns for a total of 296 events. Therefore, the Class MC component design value of 500 cycles for startups and shutdowns remains well above the projected value. Also, the one cycle allowed for a design basis accident is a value assumed in the design for a faulted condition for the life of the plant, whether that is 40 years or 60 years. Hence, the performance of these components will not be impacted by extending the life of the plant to 60 years.

The TLAA associated with thermal cycles on the ASME Class MC components have been dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

3.14.3.5.2 Downcomer Vents and Safety Relief Valve Discharge Piping

Downcomer vents and safety relief valve (SRV) discharge piping penetrate the drywell / suppression pool diaphragm slab with the purpose of transporting steam and non-condensable gases to the suppression pool from the reactor and from the drywell during SRV lifts and under accident conditions. To ensure the integrity of the downcomers and SRV discharge piping for the original 40-year life of the plant, extensive analyses were performed. These analyses satisfy the definition for TLAA.

The significant area analyzed for the downcomers in the suppression pool air space was the downcomer penetration through the diaphragm slab. Structural analyses of all the SRV discharge lines from the diaphragm slab penetration to the quencher were performed, including flued head connections, elbows, and three-way restraint attachments.

The design rules, as set forth in the ASME Section III, Subsection NB were used for the fatigue assessment. The downcomers and SRV discharge lines were analyzed for the appropriate load combinations and their associated number of cycles. The combined stresses and corresponding equivalent stress cycles were computed to obtain the fatigue usage factors in accordance with the equations of Subsection NB-3600 of the ASME Code. The maximum cumulative usage factors for the downcomers and SRV discharge lines for the 40-year plant lifetime were determined from these analyses.

The minimum number of SRV actuations assumed in any of the fatigue analyses was 1100. The projected number of events for 60 years is less than the number assumed in the design basis (40 year) analysis. Therefore, the design basis analysis remains valid for the period of extended operation

The TLAA associated with stress cycles on the downcomer vents and safety relief valve discharge piping have been dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

3.14.3.5.3 Safety Relief Valve Quenchers

Quenchers provide proper dispersion of reactor steam into the suppression pool upon lifts of SRVs and discharge of the steam through the SRV discharge piping.

Analyses for fatigue of the quenchers satisfy 10 CFR 54.3 criteria as TLAA. Fatigue evaluations for the original 40-year life of the plant list 7000 cycles as the expected number of cycles for each quencher component analyzed. The evaluations calculate the number of allowable cycles for the components and give the expected CUF for each analysis.

Since a quencher can experience up to seven cycles each time its associated SRV actuates (lifts), the worst case number of cycles is seven times the number of actuations projected for 40 years and for 60 years. These projected cycles were compared with analysis data results.

The design cycles exceed the number of cycles projected to 60 years for all components which were analyzed for the quencher. Therefore, the CUFs calculated in the fatigue evaluation remain valid for the period of extended operation.

The TLAA associated with stress cycles on the safety relief valve quenchers have been dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

3.14.3.6 Other Plant-Specific Time-Limited Aging Analyses

3.14.3.6.1 Main Steam Flow Restrictor Erosion Analyses

A flow restrictor is incorporated in each main steam line to limit flow to 200 percent of rated flow in the event that a main steam line ruptures outside containment. Erosion of a flow restrictor is a safety concern since it could impair the ability of the flow restrictor to limit vessel blowdown following a main steam line break.

FSAR Section 5.4.4.4 discusses an evaluation of the effect of potential erosion of main steam line flow restrictors on radiological dose resulting from a main steam line break accident.

Operating for another 20 years will allow further erosion and, therefore, increased opening in the throat of the flow restrictors. The erosion can be linearly extrapolated from 40 to 60 years. This is conservative since the rate of erosion would be expected to decrease as the throat area of the restrictor increased due to erosion. (It has been determined that operation at Extended Power Uprate conditions will not significantly affect the erosion rate.) Therefore, it can be concluded that erosion for the 20 years of extended operation will be no more than half the erosion for the first 40 years, and the corresponding increase in steam flow will be no more than half of the increase in steam flow due to erosion at the end of 40 years (namely, 5 percent). This means that by the end of 60 years, the increase in flow compared to flow at the beginning of life will be no more than 7.8 percent. Therefore, the released dose for the accident case at 60 years would be no more than a 7.8 percent increase. Such an increase in dose over the analyzed case remains within regulatory limits, as indicated in FSAR Section 15.6.4.5.3.

Hence, the performance of the main steam line flow restrictors is not significantly impacted by the additional erosion during the period of extended operation.

3.14.3.6.2 High Energy Line Break Cumulative Fatigue Usage Factors

High energy line breaks have been postulated and analyzed for potential effects on surrounding equipment and systems. FSAR Section 3.6 provides criteria for determining break locations and types of breaks that could occur, descriptions of analysis methodologies, and results for significant attached piping showing where breaks could develop and where restraints were to be installed. Cumulative fatigue usage factors (CUFs) 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 postulated breaks are in piping for systems important to safety and integrity of the reactor coolant pressure boundary. The restraints designed for these potential breaks are significant for protection of systems and equipment important to plant safety. Therefore, the CUF calculations used in the selection of postulated high energy line break locations are TLAA.

Since these breaks are postulated to occur only once in the lifetime of the plants and restraints were installed appropriately to mitigate these potential breaks, the results of analyses for the potential breaks and the restraints installed in the plants 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 that 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.

Presently, SSES utilizes the EPRI FatiguePro software to monitor fatigue at the critical bounding locations of piping systems in the plant. The SSES Fatigue Monitoring Program will identify when piping systems are approaching their original 40-year design CUFs. Prior to any piping system exceeding its' original maximum design CUF, the pertinent design calculations for the affected system will be reviewed to determine if any additional locations should be designated as postulated high energy line breaks, under the original criteria of FSAR Section 3.6. If other locations are determined to require consideration as postulated break locations, appropriate actions will be taken to address the new break locations.

The Fatigue Monitoring Program is credited for managing the effects of fatigue during the period of extended operation. Therefore, the TLAA associated with high energy line break cumulative fatigue have been dispositioned in accordance with10CFR54.21(c)(1)(iii).

3.14.3.6.3 Core Plate Rim Hold-Down Bolts

The NRC safety evaluation report that references BWRVIP-25, "BWR Core Plate Inspection and Flaw Evaluation Guidelines," for license renewal identifies loss of preload on the core plate rim hold-down bolts as one of the TLAA that must be addressed by applicants seeking license renewal.

SSES will address the loss of preload on the core plate rim hold-down bolts by conforming to BWRVIP-25, Revision 1 (as approved by NRC SER ML19290G703). The evaluation will be completed no less than two years prior to the period of extended operation.

3.14.3.6.4 Irradiation Assisted Stress Corrosion Cracking (IASCC)

Austenitic stainless steel reactor internal components exposed to a neutron fluence of greater than 5 x 10^{20} n/cm^2 (E > 1 MeV) are susceptible to irradiation assisted stress corrosion cracking (IASCC) in the BWR environment. Analyses were performed to determine neutron fluence for extended power uprate (EPU) conditions and for extended operation out to 54 effective full power years (EFPY). The projected fluence values are used to identify the components that exceed the threshold fluence for IASCC.

The following reactor internal components have been identified as being susceptible to IASCC for the period of extended operation for SSES Units 1 and 2:

a) Top Guide

b) Core Shroud

c) In-Core Flux Monitoring Dry Tubes

d) Core Plate

The components identified as being susceptible to IASCC require aging management to identify and address potential degradation (crack initiation and growth) prior to any loss of intended function.

All identified components have been evaluated for IASCC by the BWRVIP, as described in the inspection and evaluation guideline reports for each component: BWRVIP-26-A for the Top Guide; BWRVIP-76 for the Core Shroud; BWRVIP-47-A for the In-Core Flux Monitoring Dry Tubes; and BWRVIP-25, Revision 1 (as approved by NRC SER ML19290G703), for the Core Plate. The inspection and evaluation guidelines of the identified BWRVIP reports are implemented by the BWR Vessel Internals Program for SSES.

3.14.4 LICENSE RENEWAL COMMITMENT LIST

The listing of commitments identified for SSES license renewal is provided in Table 3.14-1. These commitments are tracked within SSES's regulatory commitment management program.

3.14.5 NEWLY IDENTIFIED ITEMS (10 CFR 54.37 (b))

After the renewed license is issued, the FSAR update required by 10 CFR 50.71(e) must include any systems, structures, and components newly identified that would have been subject to an aging management review or evaluation of time-limited aging analyses in accordance with 10 CFR 54.21. This FSAR update must describe how the effects of aging will be managed such that the intended function(s) in 10 CFR 54.4(b) will be effectively maintained during the period of extended operation. These newly identified items are listed in Table 3.14-2.

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
1. Inservice Inspection (ISI) Program	Existing program is credited.	3.14.2.23	Ongoing
2. BWR Water Chemistry Program	Existing program is credited.	3.14.2.11	Ongoing
3. Reactor Head Closure Studs Program	Existing program is credited.	3.14.2.40	Ongoing
4. BWR Vessel ID Attachment Welds Program	Existing program is credited.	3.14.2.9	Ongoing
5. BWR Feedwater Nozzle Program	Existing program is credited.	3.14.2.6	Ongoing
6. BWR CRD Return Line Nozzle Program	Existing program is credited. • SSES will implement weld overlay repairs in accordance with ASME Section XI and NRC-approved Code Cases. If no NRC-approved Code Case exists for the weld overlay, SSES will obtain NRC approval prior to implementing the repair in accordance with 10 CFR 50.55a.	3.14.2.5	Ongoing
7. BWR Stress Corrosion Cracking (SCC) Program	Existing program is credited.	3.14.2.8	Ongoing

Table 3.14-1

SSES License Renewal Commitments

Item Number		Commitment	FSAR Description Location	Enhancement or Implementation Schedule
8.	BWR Penetrations Program	Existing program is credited.	3.14.2.7	Ongoing
9.	BWR Vessel Internals Program	Existing program is credited. • SSES will continue to perform inspections on at least 10% of the top guide grid beam cells containing control rod drives/blades every twelve years during the period of extended operation. Inspections on at least 5% of the top guide locations will be performed within the first six years of each twelve year interval. The top guide locations to be inspected are those subject to neutron fluence levels that exceed the IASCC threshold of 5.0E+20 n/cm^2. The inspections will be performed using the enhanced visual inspection technique, EVT-1.	3.14.2.10	Ongoing
10.	Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steel (CASS) Program	Program is new. The new program for SSES will be consistent with the program described in NUREG-1801 Section XI.M13, Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steel (CASS) Program. The SSES program will identify susceptible components, evaluate those components to determine their susceptibility to loss of fracture toughness, and examine those components that are evaluated to be susceptible.	3.14.2.48	Prior to the period of extended operation.
11.	Flow-Accelerated Corrosion (FAC) Program	Existing program is credited.	3.14.2.20	Ongoing
12.	Bolting Integrity Program	Existing program is credited with the following enhancement: • Include specific precautions against the use of sulfur (sulfide)	3.14.2.2	Prior to the period of extended operation.

Table 3.14-1 SSES License Renewal Commitments			
Item Number	**Commitment**	**FSAR Description Location**	**Enhancement or Implementation Schedule**
	containing compounds as a lubricant for bolted connections.		
13. Piping Corrosion Program	Existing program is credited with the following enhancements: • Include the Standby Gas Treatment System loop seals within the scope of the program. • Incorporate performance, documentation and trending of opportunistic visual inspections (during normal maintenance/repair activities) in addition to existing Piping Corrosion Program inspections.	3.14.2.38	Prior to the period of extended operation.
14. Crane Inspection Program	Existing program is credited.	3.14.2.17	Ongoing
15. Fire Protection Program	Existing program is credited.	3.14.2.18	Ongoing
16. Buried Piping Surveillance Program	Program is new. The scope of the Buried Piping Surveillance Program includes only the portions of the buried piping in the Residual Heat Removal Service Water (RHRSW) and Emergency Service Water (ESW) common return header known to have damaged coatings. The program is credited for managing loss of material due to crevice, general, and pitting corrosion and microbiologically influenced corrosion (MIC) for buried steel piping components with damaged coatings.	3.14.2.4	Prior to the period of extended operation.

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
17. Condensate and Refueling Water Storage Tanks Inspection	Program is a new one-time inspection. The scope of the Condensate and Refueling Water Storage Tanks Inspection includes the base (bottom surface and foundation pad interface) of the Condensate Storage Tanks (CSTs) and Refueling Water Storage Tank (RWST) that are in the scope of license renewal and included in the Condensate Storage and Transfer and the Refueling Water Storage and Transfer systems. An appropriate combination of volumetric (including thickness measurement) and visual examinations will be conducted, for a unit's CST (or RWST), to detect evidence of a loss of material due to crevice, general, or pitting corrosion or to confirm a lack thereof. Results will be applied to the other unit's tank(s) based on engineering evaluation.	3.14.2.14	Within the 10-year period prior to the period of extended operation.
18. Reactor Vessel Surveillance Program	Existing program is credited with the following enhancement: • Address the additional requirements specified in the NRC safety evaluation dated March 1, 2006, for BWRVIP-116. The program will include a requirement that, if a standby capsule is removed from either of the SSES Unit 1 or Unit 2 reactor vessels without the intent to test it, the capsule will be stored in a manner which maintains it in a condition which would permit its future use, including during the period of extended operation if necessary.	3.14.2.41	Prior to the period of extended operation.

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
19. Chemistry Program Effectiveness Inspection	Program is a new one-time inspection. The Chemistry Program Effectiveness Inspection includes the internal surfaces of aluminum, copper and copper alloy, carbon and low alloy steel, cast iron, stainless steel, and nickel alloy components in systems that contain treated water or fuel oil. A representative sample of components in low flow and stagnant areas (i.e., locations that are isolated from the flow stream and possibly prone to gradual accumulation/concentration of contaminants) will be examined for evidence of a loss of material (due to crevice, galvanic, general, or pitting corrosion or to erosion, and to MIC in fuel oil), or to confirm a lack thereof, and the results applied to the rest of the system(s) based on engineering evaluation.	3.14.2.12	Within the 10-year period prior to the period of extended operation.
20. Cooling Units Inspection	Program is a new one-time inspection. The Cooling Units Inspection activities focus on a representative sample population of subject components at susceptible locations, to be defined in the implementing documents. These inspection activities provide symptomatic evidence of cracking, loss of material, or reduction in heat transfer at all other susceptible locations due to the similarities in materials and environmental conditions.	3.14.2.16	Within the 10-year period prior to the period of extended operation.
21. Heat Exchanger Inspection	Program is a new one-time inspection. The Heat Exchanger Inspection detects and characterizes conditions to determine whether, and to what extent a loss of heat transfer due to fouling is occurring (or likely to occur) for heat exchangers within the scope of license renewal. The Heat Exchanger Inspection is also credited for managing cracking due to stress corrosion cracking / inter-granular attack in the treated water (internal) environment of the admiralty brass tubes.	3.14.2.22	Within the 10-year period prior to the period of extended operation.
22. Not Used		3.14.2.30	

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
23. Monitoring and Collection System Inspection	Program is a new one-time inspection. The scope of the Monitoring and Collection System Inspection includes the internal surfaces of subject carbon steel (and low alloy steel) and cast iron piping and valve bodies that are exposed to potentially radioactive drainage water (untreated water) and potentially other contaminants/fluids during normal plant operations. A representative sample of components in the system, to be defined in the implementing documents, and to include containment isolation piping and/or valve bodies, will be examined for evidence of a loss of material (due to crevice, general, or pitting corrosion or to MIC), or to confirm a lack thereof, and the results applied to the rest of the system based on engineering evaluation.	3.14.2.33	Within the 10-year period prior to the period of extended operation.
24. Supplemental Piping/Tank Inspection	Program is a new one-time inspection. The Supplemental Piping/Tank Inspection is credited for managing loss of material due to crevice and pitting corrosion on carbon steel surfaces at air-water interfaces. The inspection is also credited for managing loss of material due to microbiologically influenced corrosion (MIC) at the air-water interface with the mist eliminator loop seal, which is filled with raw water from the Service Water System, and galvanic corrosion at points of contact between the mist eliminator housing and the SGTS filter enclosure, where condensation and water pooling may occur. Additionally, the Supplemental Piping/Tank Inspection detects and characterizes whether, and to what extent, a loss of material due to crevice and pitting corrosion is occurring (or is likely to occur) for stainless steel surfaces at air-water interfaces. The Supplemental Piping/Tank Inspection also detects and characterizes loss of material due to crevice, galvanic, general, and pitting corrosion on internal carbon steel surfaces within the scram discharge volume (piping and valve bodies) of the Control Rod Drive Hydraulic System, within the air	3.14.2.46	Within the 10-year period prior to the period of extended operation.

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
	space of the condensate storage tanks and within the Diesel Generator starting air receiver tanks and E diesel compressor skid air receiver tanks to determine whether, and to what extent, degradation is occurring (or is likely to occur). In addition, the Supplemental Piping/Tank Inspection is credited to detect and characterize loss of material due to general, crevice, and pitting corrosion on the internal surfaces of carbon steel and cast iron diesel exhaust piping, piping components, and turbocharger casings. The inspection is also credited to detect and characterize cracking and loss of material due to crevice and pitting corrosion on the internal surfaces of stainless steel diesel exhaust piping components.		
25. Selective Leaching Inspection	Program is a new one-time inspection. The Selective Leaching Inspection detects and characterizes conditions to determine whether, and to what extent a loss of material due to selective leaching is occurring (or likely to occur) for susceptible components including piping and tubing, valve bodies, pump and turbocharger casings, heat exchanger, cooler, and chiller components, hydrants, sprinkler heads, strainers, level gauges, orifices, and heater sheaths. The components within the scope of the program are formed of cast iron, brass, bronze, and copper alloy materials. The components are subject to raw water, treated water, groundwater (buried), indoor air with condensation, outdoor air, and fuel oil environments. The components within the scope of this program are located in twenty-six different plant systems.	3.14.2.43	Within the 10-year period prior to the period of extended operation.

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
26. Buried Piping and Tanks Inspection Program	Program is new. The scope of the Buried Piping and Tanks Inspection Program includes buried components that are within the scope of license renewal for SSES. The program is credited for managing loss of material due to crevice, general, and pitting corrosion and microbiologically influenced corrosion (MIC) for buried steel piping components. In addition, the program is credited with managing loss of material for buried stainless steel piping components. The Buried Piping and Tanks Inspection Program is also credited for managing loss of material due to crevice, general, and pitting corrosion and microbiologically influenced corrosion (MIC) for buried steel tanks in the Diesel Fuel Oil System.	3.14.2.3	Prior to the period of extended operation.
27. Small-Bore Class 1 Piping Inspection	Program is a new one-time inspection. The SSES program will include measures to verify that cracking is not occurring in Class 1 small-bore piping, thereby validating the effectiveness of the Chemistry Program to mitigate cracking and confirming that no additional aging management programs are needed for the period of extended operation.	3.14.2.44	Within the 10-year period prior to the period of extended operation.

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
28. System Walkdown Program	Existing program is credited with the following enhancements: • The governing procedure for the System Walkdown Program must be revised to add the listing of systems crediting the program for license renewal and to explicitly include inspection of other metals, copper alloy and stainless steel. ○ It may be determined by engineering evaluation that these components do not require monitoring every two weeks, and the basis for a different walkdown frequency must be documented on the appropriate procedure form. • The governing procedure for the System Walkdown Program must be enhanced to address the license renewal requirement for opportunistic inspections of normally inaccessible components (e.g., those that are insulated), and those that are accessible only during refueling outages. For underground vaults/pits/manholes, an initial sample of at least one vault/pit/manhole from each grouping of components with identical material and environment combinations will be inspected prior to entering the period of extended operation. A representative sample of the entire population will be inspected within the first 6 years of the period of extended operation. Results of the inspection activities that require further engineering evaluation/resolution (e.g., sample expansion and inspection frequency changes if degradation is detected), if any, will be evaluated using the SSES corrective action process. • The governing procedure for the System Walkdown Program will be enhanced to include a visual and ultrasonic inspection of the external surfaces of piping passing into structures through penetrations (underground piping) for those penetrations with a history of leakage. These inspections will be focused on penetrations that are leaking at that time and will include a	3.14.2.47	Prior to the period of extended operation.

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
	representative population of each material, environment combination from those piping systems within the scope of license renewal (which includes those for the RHRSW, ESW, and Fire Protection systems) that enter structures below grade. • A routine activity to supplement the existing plant program will be generated, and based at least in part on EPRI 1007933, "Aging Assessment Field Guide," to inspect elastomers and polymers for cracking and/or change in material properties. ○ Evidence of surface degradation, such as cracking or discoloration, as well as physical manipulation and/or prodding, will be used as a measure of the material condition. ○ A representative sample will be determined by engineering evaluation with a focus on components considered to be most susceptible to aging, such as due to their time in service, the severity of conditions during normal plant operations, and any pertinent design margins.		
29. Inservice Inspection (ISI) Program – IWE	Existing program is credited.	3.14.2.24	Ongoing
30. Inservice Inspection (ISI) Program – IWF	Existing program is credited.	3.14.2.25	Ongoing
31. Inservice Inspection (ISI) Program - IWL	Existing program is credited.	3.14.2.26	Ongoing

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
32. Containment Leakage Rate Test Program	Existing program is credited.	3.14.2.15	Ongoing
33. Masonry Wall Program	Existing program is credited with the following enhancement: • Specify that for each masonry wall, the extent of observed masonry cracking and/or degradation of steel edge supports/bracing is evaluated to ensure that the current evaluation basis is still valid. Corrective action is required if the extent of masonry cracking and steel degradation is sufficient to invalidate the evaluation basis.	3.14.2.31	Prior to the period of extended operation.
34. Structures Monitoring Program	Existing program is credited with the following enhancements: • Include additional structures requiring aging management for license renewal to the scope of the inspections. • Specify that if a below grade structural wall or structural component becomes accessible through excavation; a follow-up action is initiated for the responsible engineer to inspect the exposed surfaces for age-related degradation. • Clarify “structural component” for inspection includes each of the component types identified as requiring aging management. • Include degradation mechanisms for elastomer and earthen embankment inspection. • Include RG 1.127 inspection elements for water-control structure. • Specify that the responsible engineer shall review site	3.14.2.45	Prior to the period of extended operation.

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SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
	groundwater and raw water pH, chlorides, and sulfates results prior to inspection to validate that the below-grade or raw water environment remains non-aggressive during the period of extended operation. • Specify that for each masonry wall, the extent of observed masonry cracking and/or degradation of steel edge supports/bracing is evaluated to ensure that the current evaluation basis is still valid. Corrective action is required if the extent of masonry cracking and steel degradation is sufficient to invalidate the evaluation basis. • Include additional direction for quantifying, monitoring and trending of inspection results; Include additional guidance for inspection reporting, data collection and documentation; Specify acceptance criteria and critical parameters to monitor degradation and to trigger level of inspection and initiation of corrective action; and provide better alignment with referenced Industry codes, standards and guidelines. • Include specific qualification requirements for the inspector.		

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
35. RG 1.127 Water-Control Structures Inspection	Existing program is credited with the following enhancements: • Add the Spray Pond (including concrete liner, emergency spillway, riser encasements and earthen embankments) to its scope for inspection. • Include RG 1.127 Revision 1 Section C.2 inspection elements and degradation mechanisms for water-control structure inspection. • Include acceptance criteria as delineated in NUREG-1801 Section XI.S7 for water-control structures. Evaluation criteria provided in Chapter 5 of ACI 349.3R-96 provides acceptance criteria (including quantitative criteria) for determining the adequacy of observed aging effects and specifies criteria for further evaluation.	3.14.2.42	Prior to the period of extended operation.
36. Non-EQ Electrical Cables and Connections Visual Inspection Program	Program is new. The Non-EQ Electrical Cables and Connections Visual Inspection Program is credited with detecting aging effects from adverse localized environments in non-EQ cables and connections at SSES. The program is applicable to non-EQ cables and connections found in the Reactor Buildings, Circulating Water Pumphouse and Water Treatment Building, Control Structure, Diesel Generator Buildings, Turbine Building, Engineered Safeguards Service Water Pumphouse, and various yard structures (manholes, duct banks, valve vaults, instrument pits, etc.). This program is also applicable to the cables and connections within the scope of license renewal located in the yard areas and control cubicles of the T10 230 kV Switchyard, the 500 kV Switchyard, and the 230 kV Switchyard.	3.14.2.35	Prior to the period of extended operation.

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SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
37. Non-EQ Cables and Connections Used in Low-Current Instrumentation Circuits Program	Program is new. The Non-EQ Cables and Connections Used in Low-Current Instrumentation Circuits Program is credited with identifying aging effects for sensitive, high-voltage, low-current signal applications that are in-scope for license renewal at SSES. These sensitive circuits are potentially subject to reduction in insulation resistance (IR) when found in adverse localized environments.	3.14.2.34	Prior to the period of extended operation.
38. Non-EQ Inaccessible Medium-Voltage Cables Program	Program is new. The Non-EQ Inaccessible Medium-Voltage Cables Program involves two parts: first, the actions to inspect the applicable plant manholes (and to drain them, if necessary) on a periodic basis; and second, the development of a testing program to confirm that the conductor insulation on the applicable cables is not degrading. This program applies to six cables associated with the offsite power supply for SSES. These are the only inaccessible medium-voltage cables at SSES that are within the scope of license renewal and are exposed to significant moisture simultaneously with significant voltage.	3.14.2.36	Prior to the period of extended operation.
39. Metal-Enclosed Bus Inspection Program	Program is new. The Metal-Enclosed Bus Inspection Program is credited with detecting aging effects for in-scope metal-enclosed bus at SSES. The applicable components for the metal-enclosed bus will be listed in the program implementing document(s), with their locations specified, as appropriate. The in-scope bus is limited to non-segregated metal-enclosed bus in the 13.8 kV and 4 kV electrical systems associated with the off-site power supply at SSES.	3.14.2.32	Prior to the period of extended operation.

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SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
40. Area-Based NSAS Inspection	Program is a new one-time inspection. The Area-Based NSAS Inspection includes confirming the environmental and/or internal surfaces conditions of subject nonsafety-related carbon steel (includes low alloy steel), cast iron, copper alloy and stainless steel components in systems that (frequently or continuously during normal plant operations) contain raw water, potable water, non-radioactive equipment/area drainage water, or in some select cases, treated water. The program is plant-specific.	3.14.2.1	Within the 10-year period prior to the period of extended operation.
41. Leak Chase Channel Monitoring Activities	The existing program is credited. The program is plant-specific.	3.14.2.27	Ongoing
42. Preventive Maintenance Activities – RCIC/HPCI Turbine Casings	Existing program is credited with the following enhancements: • Include a specific step to perform a visual inspection of the RCIC turbine casing. • Add requirements to have inspections performed by qualified personnel using VT-3 or equivalent inspection methods, and to document and trend inspection results. • Establish specific acceptance criteria for inspection results. The program is plant-specific.	3.14.2.39	Prior to the period of extended operation.
43. Fatigue Monitoring Program	Existing program is credited with the following enhancements: • Provisions will be made in the Fatigue Monitoring Program to validate that components which have satisfied ASME Section III, Paragraph N-415.1 requirements (i.e., RPV nozzles N6A, N6B, and	3.14.2.49	Prior to the period of extended operation.

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SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
	N7) continue to satisfy these requirements prior to and during the period of extended operation, thereby allowing fatigue to be continued to be addressed under N-415.1. • The Fatigue Monitoring Program will be enhanced to ensure that the fatigue usage at all monitored locations, including those locations that account for the effect of the reactor water environment, is managed such that an adequate margin against fatigue cracking is maintained. SSES will implement one or more of the following actions, if fatigue usage at a monitored location, including any location that accounts for the effect of the reactor water environment, is projected to reach the design basis limit prior to the end of the period of extended operation: 1. Further refinement of the fatigue analyses to lower the CUFs to less than the allowable; 2. Repair of the affected components; 3. Replacement of the affected components; 4. Management by an inspection program that has been reviewed and approved by the NRC. • The Fatigue Monitoring Program will be enhanced to include the review of Class 1 valve fatigue analyses and other fatigue-related TLAA, such as flued head analyses and high energy line break evaluations, when sufficient fatigue accumulation has occurred, to determine if additional actions are required to address fatigue-related concerns. • The Fatigue Monitoring Program will be enhanced to include fatigue monitoring of the additional locations required to bound the limiting		

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
	locations applicable to SSES, as identified in NUREG/CR-6260. • The Fatigue Monitoring Program will be enhanced to establish monitoring criteria to ensure that the fatigue usage at all monitored locations, including those locations that account for the effect of the reactor water environment, is managed such that design basis limits are not exceeded during the period of extended operation. The Fatigue Monitoring Program will define specific fatigue usage values for all monitored locations that, if reached, will require further action. These fatigue usage values shall be conservatively set to values that will allow for not less than 4 years of additional plant operation before the actual fatigue usage at any location would reach the design basis limit. Upon reaching the defined usage at a location, the Fatigue Monitoring Program will require an action request to be generated. The action request will require further engineering evaluation to resolve the issue.		
44. Environmental Qualification (EQ) Program	Existing program is credited. For those EQ components that do not show a minimum 60-year life, the EQ Program will ensure qualified life is not exceeded by directing refurbishment, replacement, or reanalysis to extend the qualification.	3.14.3.4	Ongoing
45. Closed Cooling Water Chemistry Program	Existing program is credited.	3.14.2.13	Ongoing.

Table 3.14-1
SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
46. Fire Water System Program	Existing program is credited with the following enhancements: • The Fire Water System Program will be revised to incorporate sprinkler head sampling/replacements, in accordance with NFPA 25. • The Fire Water System Program will be revised to incorporate ultrasonic testing of representative above ground portions of water suppression piping that are exposed to water but which do not normally experience flow, are associated with a dry-pipe sprinkler system and may contain stagnant water, or is pre-action or deluge piping that is normally dry but may have been wetted and not completely dried. • Perform at least one visual inspection (opportunistic or focused) of the internal surface of buried fire water piping, within the 10 year period prior to the period of extended operation. • Perform at least one inspection per year of 'wet' fire protection piping for wall thickness and pipe blockage, if no opportunistic inspection has been completed.	3.14.2.19	Prior to the period of extended operation.
47. Fuel Oil Chemistry Program	Existing program is credited.	3.14.2.21	Ongoing

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
48. Lubricating Oil Analysis Program	Existing program is credited with the following enhancements: • The Lubricating Oil Analysis Program will be enhanced to include sampling of the lubricating oil from the Control Structure Chiller and Diesel Engine Driven Fire Pumps when the oil is changed. The oil will be tested for water and, excluding the Diesel Engine Driven Fire Pumps, for particle count. The Diesel Engine Driven Fire Pumps will be tested via direct read ferrography and spectrochemical testing instead of particle counting. • The Lubricating Oil analysis Program will be revised to include sampling of the lubricating oil from the Reactor Building Chiller when the oil is changed. The oil will be tested for water and particle count.	3.14.2.28	Prior to the period of extended operation.
49. Lubricating Oil Inspection	Program is a new one-time inspection. The Lubricating Oil Inspection detects and characterizes the condition of materials in systems and components for which the Lubricating Oil Analysis Program is credited with aging management. The inspection provides direct evidence as to whether, and to what extent, a loss of material or a reduction in heat transfer due to fouling has occurred.	3.14.2.29	Within the 10-year period prior to the period of extended operation.
50. Non-EQ Electrical Cable Connections Program	Program is new. The Non-EQ Electrical Cable Connections Program manages the aging for the metallic parts of non-EQ electrical cable connections within the scope of license renewal. The program addresses cable connections that are used to connect cable conductors to other cables or electrical devices. Aging management for the metallic parts of the non-EQ electrical cable connections that are subject to aging stressors will be provided by testing.	3.14.2.37	Prior to the period of extended operation.

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SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
51. New P-T Curves	Revised Pressure-Temperature (P-T) limits will be submitted to the NRC for approval when necessary to comply with 10 CFR 50 Appendix G.	3.14.3.1.4	Ongoing
52. OE Review at EPU Conditions	Perform an Operating Experience (OE) review for the period of operation at EPU conditions and its impact on aging management programs for systems, structures and components (SSCs).	---------	Prior to the period of extended operation.
53. Incorporate FSAR Supplement	Incorporate FSAR Supplement into the SSES FSAR as required by 10 CFR 54.21(d).	3.14.1	Following issuance of the renewed operating licenses.
54. Re-apply for relief request	SSES will process a relief request for circumferential vessel shell weld volumetric examinations for the period of extended operation.	3.14.3.1.5	Prior to the period of extended operation.
55. Core Plate Hold Down Bolts	SSES will address the loss of preload on the core plate rim hold-down bolts by conforming to BWRVIP-25, Revision 1 (as approved by NRC SER ML19290G703). The evaluation will be completed no less than two years prior to the period of extended operation.	3.14.3.6.3	Prior to the period of extended operation.
56. BWRVIP-76	SSES will address any future conditions, requirements, or limitations imposed by the NRC's safety evaluation for license renewal for BWRVIP-76.	3.14.3.6.4	Prior to the period of extended operation.
57. Preventative Maintenance Activities-Main Turbine Casing	Existing program is credited with the following enhancement: • Specify that the inspection of the high pressure turbine shell will consist of a visual inspection (VT-3 or equivalent) of accessible surfaces and an ultrasonic examination of selected locations for wall thickness. The program is plant specific.	3.14.2.50	Prior to the period of extended operation.

Table 3.14-1

SSES License Renewal Commitments

Item Number	Commitment	FSAR Description Location	Enhancement or Implementation Schedule
58. Activities in Response to NRC Generic Letter 88-14	Activities credited in the SSES response to NRC Generic Letter 88-14 will be continued throughout the period of extended operation.	---------	Ongoing
59. Fuse Holders Program	Program is new. The Fuse Holders Program is credited with identifying increased connection resistance between the fuse holder metallic clamp and fuse due to fatigue of the metallic clamp. The program provides for periodic inspection of fuse holder clamps within the scope of license renewal that are not in enclosures containing active components and whose fuses are scheduled for removal once every 12 months, or more frequently.	3.14.2.51	Prior to the period of extended operation.
60. Activities in Response to NRC Concerns Regarding Fatigue Analyses	SSES will either (1) implement fatigue monitoring software that satisfactorily addresses all issues raised in Regulatory Information Summary (RIS) 2008-30, “Fatigue Analysis of Nuclear Power Plant Components”, or (2) perform a confirmatory ASME Code, Section III fatigue evaluation for the SBF-monitored locations to justify the existing FatiguePro methodology used at SSES Units 1 and 2.	-----------	Prior to the period of extended operation.
61. Boral Coupon Testing	Spent fuel pool Boral coupon testing will be continued in the period of extended operation with one set of coupons being tested during the tenth or eleventh year after Unit 1 enters the period of extended operation. SSES FSAR section 9.1.2.3.3 Inservice Inspection will be revised to identify the coupon testing schedule during the period of extended operation. SSES FSAR section 9.1.2.3.3.2 Test Coupon Inspection will be revised to require neutron attenuation testing as part of the inspection of test coupons removed from the spent fuel pool.	-----------	Revise the FSAR prior to the period of extended operation, with coupon testing ongoing as indicated.

Table 3.14-2

Newly Identified Structures, Systems, and Components (SSC)

No.	FSAR Revision No.	SSC & Description	Aging Management Review Conclusion	Aging Management Program
1	69	Backup Fire Protection System: • The original Backup Fire Protection System was installed in the plant at the time of the license renewal review, in accordance with the CLB at the time. The Backup Fire Protection System should have been included in the scope of license renewal per 10 CFR 54.4(a); however, the system was not identified as in scope until after issuance of the renewed license. The Backup Fire Protection System (both the original and 2017 modification) is credited to perform a function meeting the requirements of 10 CFR 50.48.† † The backup diesel fire pump, backup jockey pump and other components were replaced with a modified design after the renewed license was issued and are therefore not subject to the provisions of 10 CFR 54.37(b). As such, these SSCs are not in LR scope.	Components added to scope because of this change: • Pipe • Valves • Well Water Storage Tank Materials of Construction: • Carbon Steel • Cast Iron Environments: • Buried • Outdoor (uncontrolled) • Sheltered Aging effects for the material/environment combinations: • Loss of material in carbon steel/cast iron in Potable Water, Outdoor, and Buried environments	There are no changes required to any of the AMPs for fire protection as identified in LRA Table 3.3.2-13 for aging management of the newly identified SSCs. The material/environment combination is the same as the material/environment combination previously evaluated for the fire protection SSCs in the License Renewal Application, therefore, the aging effects and aging management programs previously credited are being used to manage the newly added backup fire protection system SSCs.