Appendix A

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APPENDIX A

SUPPLEMENT AGING MANAGEMENT PROGRAMS AND ACTIVITIES CREDITED FOR COLUMBIA LICENSE RENEWAL

A.0 FINAL SAFETY ANALYSIS REPORT SUPPLEMENT

A.1 INTRODUCTION

This appendix provides the information submitted for the Final Safety Analysis Report (FSAR) Supplement as required by 10 CFR 54.21(d) for the License Renewal Application (LRA). The programs and activities credited to manage the effects of aging are described in LRA Appendix B. Section 4 of the LRA documents the evaluations of time-limited aging analyses for the period of extended operation. LRA Section 3, Section 4, and Appendix B have been used to prepare the program and activity descriptions that are contained in this appendix.

A.2 AGING MANAGEMENT PROGRAMS AND ACTIVITIES

The license renewal integrated plant assessment identified existing and new aging management programs (AMPs) 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 aging management programs and activities identified during the integrated plant assessment. The aging management programs will be implemented prior to the period of extended operation. One-time inspections will be conducted within the 10-year period prior to beginning the period of extended operation. The aging management programs identified as necessary in association with the evaluation of time-limited aging analyses (TLAAs) are described in Section A.2.2.

Three elements of an effective aging management program that are common to each of the aging management programs are corrective actions, confirmation process, and administrative controls. These elements are included in the Operational Quality Assurance Program Description (OQAPD) for Columbia, which implements the requirements of 10 CFR 50, Appendix B.

Prior to the period of extended operation, the elements of corrective actions, confirmation process, and administrative controls in the OQAPD will be applied to required aging management programs for both safety-related and non-safety related structures and components determined to require aging management during the period of extended operation.

The existing Corrective Action Program and the Operating Experience Program ensure, through the continual review of both plant-specific and industry operating experience, that the license renewal aging management programs are effective to manage the aging effects for which they are credited. The aging management programs are either enhanced or new programs are developed when the review of operating experience indicates that the aging management programs may not be effective. For each aging management program listed in this section, operating experience is reviewed on a continuing basis.

The processes and procedures for the review of operating experience address the following points:

- All operating experience is screened for aging of long lived passive structures or components and further evaluation as applicable is performed by personnel trained in the requirements of license renewal scoping, screening, and aging management reviews (aging effects and mechanisms). The evaluation is completed and prioritized commensurate with the potential significance of the issue. Such evaluations are documented and retained in an auditable and retrievable form.

- Periodic training for system engineers, equipment operators and maintenance personnel specific to identifying aging issues.

- The License Renewal program lead is trained in the requirements of license renewal scoping, screening, and aging management reviews (aging effects and mechanisms).

- Aging management program owners are trained in the requirements of license renewal scoping, screening, and aging management reviews (aging effects and mechanisms) associated with their particular aging management program.

- When it is determined that enhancements are necessary to adequately manage the effects of aging, the enhancements are entered into and implemented consistent with the plant corrective action program or operating experience program, as applicable. Enhancements can include, as appropriate, modifications to aging management programs or the creation and implementation of new AMPs.

- Operating experience that is related to aging of long lived passive structures or components is keyword tagged “Aging.”

- The processes are adequate so as to not preclude the consideration of operating experience related to aging management. The processes appropriately gather information on all structures and components within the scope of license renewal, and their materials, environments, aging effects, and aging mechanisms. In addition, the processes include the AMPs credited for managing the effects of aging, and the activities under these AMPs (e.g., inspection methods, preventive actions, evaluation techniques, etc.).

- While the programs and procedures may specify reviews of certain sources of information, such as NRC generic communications and Institute of Nuclear Power Operations reports, they allow for any potential source of relevant plant specific or industry operating experience information.

- AMP owners review data collected by the AMPs, utilize the corrective action program for any conditions that are unsatisfactory to ensure they will be addressed and corrected, maintain required records for the program and maintain the program current and implement revisions as needed based on program results and internal or external operating experience.

- Provide guidance on sharing internal operating experience related to license renewal issues with the industry.

A.2.1 AGING MANAGEMENT PROGRAMS

A.2.1.1 ABOVEGROUND STEEL TANKS INSPECTION

The Aboveground Steel Tanks Inspection detects and characterizes the conditions on the bottom surfaces of the condensate storage tanks. The inspection provides direct evidence through volumetric examination as to whether, and to what extent, a loss of material due to corrosion has occurred in inaccessible areas (i.e., tank base and bottom surface).

The Aboveground Steel Tanks Inspection is a new inspection program 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.

A.2.1.2 AIR QUALITY SAMPLING PROGRAM

The Air Quality Sampling Program is an existing prevention and condition monitoring program that manages loss of material due to corrosion for Diesel Starting Air (DSA) components that contain compressed air through periodic sampling of the air for hydrocarbons, dewpoint, and

particulates and periodic ultrasonic inspection of the DSA System air receivers. In addition, the Air Quality Sampling Program ensures that the Control Air System remains dry and free of contaminants, such that no aging effects require management.

The Air Quality Sampling Program is supplemented by the Diesel Starting Air Inspection, which provides verification of the effectiveness of the program in mitigating the effects of aging in the DSA System dryers and the downstream piping and components (excluding the DSA System air receivers).

A.2.1.3 APPENDIX J PROGRAM

The Appendix J Program is an existing monitoring program that detects degradation of the Primary Containment and systems penetrating the Primary Containment, which are the containment shell and primary containment penetrations including (but not limited to) the personnel airlock, equipment hatch, control rod drive hatch, and drywell head. The Appendix J Program provides assurance that leakage from the Primary Containment will not exceed maximum values for containment leakage.

A.2.1.4 BOLTING INTEGRITY PROGRAM

The Bolting Integrity Program is a combination of existing activities that, in conjunction with other credited programs, address the management of aging for the bolting of mechanical components and structural connections within the scope of license renewal. The Bolting Integrity Program relies on manufacturer and vendor information and industry recommendations for the proper selection, assembly, and maintenance of bolting for pressure-retaining closures and structural connections. The Bolting Integrity Program includes, through the Inservice Inspection (ISI) Program, Inservice Inspection (ISI) Program – IWF, Structures Monitoring Program, and External Surfaces Monitoring Program, the periodic inspection of bolting for indications of degradation such as leakage, loss of material due to corrosion, loss of pre-load, and cracking due to stress corrosion cracking (SCC) and fatigue.

A.2.1.5 BURIED PIPING AND TANKS INSPECTION PROGRAM

The Buried Piping and Tanks Inspection Program manages the effects of loss of material due to corrosion on the external surfaces of metallic piping and tanks that are buried or underground. The program also manages the effects of cracking, loss of material (and loss of pre-load) for bolting that is buried. In addition, the program also verifies that aging degradation is not occurring for concrete and polymer piping that is buried. The Buried Piping and Tanks Inspection Program is a combination of a mitigation program (consisting of protective coatings. cathodic protection, and backfill quality) and a condition monitoring program (consisting of electrochemical verification of cathodic protection, confirmation of backfill quality, visual

inspections of pipe or tank external surfaces, and non-destructive evaluation of pipe or tank wall thickness as needed).

Inspection of buried and underground piping will be performed within the 10-year period prior to entering the period of extended operation. Additional inspections of buried and underground piping and buried tanks will be performed within 10 years after entering the period of extended operation, and in each 10 year period thereafter.

The Buried Piping and Tanks Inspection Program is an existing program that requires enhancement prior to the period of extended operation.

A.2.1.6 BWR FEEDWATER NOZZLE PROGRAM

The BWR Feedwater Nozzle Program is an existing program that manages cracking due to stress corrosion cracking and intergranular attack (SCC/IGA) and flaw growth of the feedwater nozzles. The BWR Feedwater Nozzle Program is in accordance with ASME Section XI and NRC augmented requirements.

The BWR Feedwater Nozzle Program consists of: (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 (2001 edition including the 2002 and 2003 Addenda) and the recommendations of General Electric report NE-523-A71-0594-A (Reference A.3-1), and (b) system modifications, as described in FSAR Section 5.3.3.1.4.5, to mitigate cracking. The program specifies periodic ultrasonic inspection of critical regions of the feedwater nozzles.

The BWR Feedwater Nozzle Program credits portions of the Inservice Inspection (ISI) Program.

A.2.1.7 BWR PENETRATIONS PROGRAM

The BWR Penetrations Program is an existing condition monitoring program that manages cracking due to SCC or intergranular stress corrosion cracking (IGSCC) of reactor vessel instrument penetrations, jet pump instrument penetrations, control rod drive penetrations, and incore instrument penetrations. The BWR Penetrations Program detects and sizes cracks in accordance with the guidelines of approved Boiling Water Reactor Vessel and Internals Project (BWRVIP) documents and the requirements of the ASME Boiler and Pressure Vessel Code, Section XI. The BWR Water Chemistry Program monitors and controls reactor coolant water chemistry in accordance with BWRVIP guidelines to ensure the long-term integrity and safe operation of the vessel components.

The program credits portions of the Inservice Inspection (ISI) Program and the BWR Vessel Internals Program.

A.2.1.8 BWR STRESS CORROSION CRACKING PROGRAM

The BWR Stress Corrosion Cracking Program is an existing condition monitoring program that manages cracking due to SCC/IGA for stainless steel and nickel alloy reactor coolant pressure boundary piping, nozzle safe ends, nozzle thermal sleeves, valve bodies, flow elements, and pump casings.

The BWR Stress Corrosion Cracking Program consists of (a) preventive measures to mitigate SCC/IGA, and (b) inspection and flaw evaluation to monitor SCC/IGA and its effects. The BWR Water Chemistry Program monitors and controls reactor coolant water chemistry in accordance with BWRVIP guidelines to ensure the long-term mitigation of SCC/IGA. The program includes the scope of the Generic Letter 88-01 program, as modified by the staff-approved BWRVIP-75 report.

The program credits portions of the Inservice Inspection (ISI) Program and the BWR Water Chemistry Program.

A.2.1.9 BWR VESSEL ID ATTACHMENT WELDS PROGRAM

The BWR Vessel ID Attachment Welds Program is an existing program that manages cracking due to SCC/IGA of the welds for internal attachments to the reactor vessel. The BWR Vessel ID Attachment Welds Program performs examinations and inspections as required by ASME Section XI, augmented by BWRVIP-48-A. These inspections include enhanced visual inspections with resolution to the guidelines in BWRVIP-03. The BWR Water Chemistry Program monitors and controls reactor coolant water chemistry in accordance with BWRVIP guidelines to ensure the long-term integrity and safe operation of the vessel internal attachment welds.

The BWR Vessel ID Attachment Welds Program credits portions of the BWR Vessel Internals Program and the Inservice Inspection (ISI) Program.

A.2.1.10 BWR VESSEL INTERNALS PROGRAM

The BWR Vessel Internals Program is an existing condition monitoring program that manages cracking due to stress corrosion cracking and irradiation assisted stress corrosion cracking (SCC/IASCC), SCC/IGA, flaw growth, and flow-induced vibration for various components and subcomponents of the reactor vessel internals. The BWR Vessel Internals Program consists of mitigation, inspection, flaw evaluation, and repair in accordance with the guidelines

of BWRVIP reports and the requirements of the ASME Boiler and Pressure Vessel Code, Section XI. The BWR Water Chemistry Program monitors and controls reactor coolant water chemistry in accordance with BWRVIP guidelines to ensure the long-term integrity and safe operation of the vessel internal components.

The BWR Vessel Internals Program credits portions of the Inservice Inspection (ISI) Program.

A.2.1.11 BWR WATER CHEMISTRY PROGRAM

The BWR Water Chemistry Program is an existing program that mitigates degradation of components that are within the scope of license renewal and contain or are exposed to treated water, treated water in the steam phase, reactor coolant, or treated water in a sodium pentaborate solution. The program manages the relevant conditions that could lead to the onset and propagation of a loss of material due to corrosion or erosion, cracking due to SCC, or reduction in heat transfer due to fouling through proper monitoring and control of chemical concentrations consistent with BWRVIP water chemistry guidelines.

The BWR Water Chemistry Program is supplemented by the Chemistry Program Effectiveness Inspection and the Heat Exchangers Inspection, to provide verification of the effectiveness of the program in managing the effects of aging. Additionally, the BWR Water Chemistry Program is supplemented by the BWR Feedwater Nozzle Program, BWR Stress Corrosion Cracking Program, BWR Penetrations Program, BWR Vessel ID Attachment Welds Program, BWR Vessel Internals Program, Inservice Inspection (ISI) Program, and Small Bore Class 1 Piping Program to provide verification of the program's effectiveness in managing the effects of aging for reactor pressure vessel, reactor vessel internals, and reactor coolant pressure boundary components.

A.2.1.12 CHEMISTRY PROGRAM EFFECTIVENESS INSPECTION

The Chemistry Program Effectiveness Inspection detects and characterizes the condition of materials in representative low flow and stagnant areas of systems with water chemistry controlled by the BWR Water Chemistry Program or the Closed Cooling Water Chemistry Program, and with fuel oil chemistry controlled by the Fuel Oil Chemistry Program. The inspection provides direct evidence as to whether, and to what extent, a loss of material due to corrosion has occurred. The inspection also determines whether cracking due to SCC of susceptible materials in susceptible locations has occurred.

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.

A.2.1.13 CLOSED COOLING WATER CHEMISTRY PROGRAM

The Closed Cooling Water Chemistry Program mitigates degradation of components that are within the scope of license renewal and contain closed cooling water. The program manages the relevant conditions that could lead to the onset and propagation of a loss of material due to corrosion or erosion, cracking due to SCC, or reduction in heat transfer due to fouling through proper monitoring and control of corrosion inhibitor concentrations consistent with EPRI closed cooling water chemistry guidelines.

The Closed Cooling Water Chemistry Program includes corrosion rate measurement in reactor building closed cooling water locations and is supplemented by the one-time Chemistry Program Effectiveness Inspection and Heat Exchangers Inspection, which provide verification of the effectiveness of the program in managing the effects of aging.

The Closed Cooling Water Chemistry Program is an existing program that requires enhancement prior to the period of extended operation.

A.2.1.14 COOLING UNITS INSPECTION PROGRAM

The Cooling Units Inspection Program manages the effect of loss of material for aluminum, steel, copper alloy, and stainless steel cooling unit components that are exposed to condensation. The inspection also manages the effects of a reduction in heat transfer due to fouling of heat exchanger tubes and fins, or cracking due to SCC of aluminum components exposed to condensation.

The Cooling Units Inspection is a new program that will be implemented via baseline inspection of a sample population followed by opportunistic inspections when components are opened for periodic maintenance, repair, and surveillance activities when surfaces are made available for inspection. These inspections ensure that the existing environmental conditions are not causing material degradation that could result in a loss of component intended function during the period of extended operation. Inspection of a sample population will be conducted within the 10-year period prior to the period of extended operation and serve as a baseline for future inspections.

A.2.1.15 CONTROL ROD DRIVE RETURN LINE NOZZLE PROGRAM

The Control Rod Drive Return Line (CRDRL) Nozzle Program is an existing mitigation and condition monitoring program that manages cracking due to flaw growth of the control rod drive return line nozzle, safe end, cap, and connecting welds. The CRDRL Nozzle Program consists of a) mitigation activities, and b) inspection, flaw evaluation, and repair in accordance with the ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWB, Table IWB

2500-1 (2001 Edition through 2003 Addenda) and the recommendations of NUREG-0619. System modifications were implemented by the original equipment manufacturer prior to initial startup to mitigate cracking. The BWR Water Chemistry Program monitors and controls reactor coolant water chemistry in accordance with BWRVIP guidelines to ensure the long-term integrity and safe operation of the critical regions of the CRDRL nozzle.

The CRDRL Nozzle Program credits portions of the Inservice Inspection (ISI) Program.

A.2.1.16 DIESEL STARTING AIR INSPECTION

The Diesel Starting Air Inspection detects and characterizes the condition of materials for the DSA System air dryers and downstream piping and components (excluding the DSA System air receivers). The inspection provides direct evidence as to whether, and to what extent, a loss of material due to corrosion has occurred.

The Diesel Starting Air 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.

A.2.1.17 DIESEL SYSTEMS INSPECTION PROGRAM

The Diesel Systems Inspection Program manages the effects of the loss of material due to corrosion and cracking due to stress corrosion cracking of materials for the interior of the steel and stainless steel exhaust piping for the Division 1, 2, and 3 diesels in the Diesel Engine Exhaust System, including the loop seal drains from the exhaust piping. The inspection provides direct evidence as to whether, and to what extent, a loss of material due to corrosion has occurred.

The Diesel Systems Inspection is a new program that will be implemented via baseline inspection of a sample population followed by opportunistic inspection when components are opened for periodic maintenance, repair, or surveillance activities when surfaces are made available for inspection. These inspections ensure that the existing environmental conditions are not causing material degradation that could result in a loss of component intended function during the period of extended operation. Inspection of a sample population will be conducted within the 10-year period prior to the period of extended operation and will serve as a baseline for future inspections.

A.2.1.18 DIESEL-DRIVEN FIRE PUMPS INSPECTION PROGRAM

The Diesel-Driven Fire Pumps Inspection Program manages the effects of the loss of material, due to corrosion or erosion, and reduction in heat transfer of the interior of the Fire Protection

System diesel engine exhaust piping, and of Fire Protection System diesel heat exchangers exposed to a raw water environment. The inspection also manages cracking due to SCC of susceptible materials.

The Diesel-Driven Fire Pumps Inspection is a new program that will be implemented via baseline inspection of a sample population followed by opportunistic inspection when components are opened for periodic maintenance, repair, or surveillance activities when surfaces are made available for inspection. These inspections ensure that the existing environmental conditions are not causing material degradation that could result in a loss of component intended function during the period of extended operation. Inspection of a sample population will be conducted within the 10-year period prior to the period of extended operation and will serve as a baseline for future inspections.

A.2.1.19 ELECTRICAL CABLES AND CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 EQ REQUIREMENTS PROGRAM

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 EQ Requirements Program is an inspection program that detects degradation of electrical cables and connections that are not environmentally qualified and are within the scope of license renewal. The program provides for periodic visual inspection of accessible, non- environmentally qualified cables and connections in order to determine if age-related degradation is occurring, particularly in plant areas with adverse localized environments. An adverse localized environment is a condition in a limited plant area that is significantly more severe than the specified design or bounding plant environment for the general area.

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 EQ Requirements Program is a new aging management program that will be implemented prior to the period of extended operation. The inspection frequency of the program will be once every 10 years, with the initial inspection to be performed prior to the period of extended operation.

A.2.1.20 ELECTRICAL CABLES AND CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 EQ REQUIREMENTS USED IN INSTRUMENTATION CIRCUITS PROGRAM

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 EQ Requirements Used in Instrumentation Circuits Program is a monitoring program that detects degradation of electrical cables and connections that are not environmentally qualified and used in circuits with sensitive, low-current applications (such as radiation monitoring and nuclear instrumentation loops). The program provides for a review of calibration records for the low-current instruments, in order to detect and identify degradation of the cable system insulation resistance. The program retains the option to perform direct cable testing.

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 EQ Requirements Used in Instrumentation Circuits Program is a new aging management program that will be implemented prior to the period of extended operation. The frequency of the program will be once every 10 years, with the initial review to be performed prior to the period of extended operation.

A.2.1.21 ELECTRICAL CABLE CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 EQ REQUIREMENTS INSPECTION

The Electrical Cable Connections Not Subject to 10 CFR 50.49 EQ Requirements Inspection detects and characterizes the material condition of metallic electrical connections within the scope of license renewal. The inspection uses thermography (augmented by contact resistance testing) to detect loose or degraded connections that lead to increased resistance for a representative sample of metallic electrical connections in various plant locations.

The Electrical Cable Connections Not Subject to 10 CFR 50.49 EQ Requirements 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.

A.2.1.22 EQ PROGRAM

Environmental qualification (EQ) analyses for electrical components with a qualified life of 40 years or greater are identified as TLAAs; therefore, the effects of aging must be addressed for license renewal.

NRC regulation 10 CFR 50.49, "Environmental Qualification of Electrical 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. The EQ Program is an existing program that implements the requirements of 10 CFR 50.49 (as further defined by the Division of Operating Reactor Guidelines, NUREG-0588, and Regulatory Guide 1.89 Revision 1).

In accordance with 10 CFR 54.21(c)(1)(iii), the EQ Program will be used to manage the effects of aging on the intended functions of the components associated with EQ TLAAs for the period of extended operation, because equipment will be replaced prior to reaching the end of its qualified life. Reanalysis addresses attributes of analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, and corrective actions if acceptance criteria are not met. Reanalysis of aging evaluations to extend the qualification of

components is performed on a routine basis pursuant to 10 CFR 50.49(e) as part of the Columbia EQ Program.

A.2.1.23 EXTERNAL SURFACES MONITORING PROGRAM

The External Surfaces Monitoring Program consists of observation and surveillance activities intended to detect degradation resulting from loss of material due to corrosion and cracking due to SCC for mechanical components, as well as hardening and loss of strength for elastomers. The External Surfaces Monitoring Program is a condition-monitoring program.

The External Surfaces Monitoring Program is an existing program that requires enhancement prior to the period of extended operation.

A.2.1.24 FATIGUE MONITORING PROGRAM

Fatigue evaluations for mechanical components are identified as TLAAs; therefore, the effects of fatigue have been addressed for license renewal.

Energy Northwest monitors fatigue of various components (including ASME Class 1 reactor coolant pressure boundary, high energy line break locations, and Primary Containment) via the Fatigue Monitoring Program, which tracks transient cycles and calculates fatigue usage. Energy Northwest has assessed the impact of the reactor coolant environment on the sample of critical components identified in NUREG/CR-6260 and other limiting components beyond those locations identified in NUREG/CR-6260. 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.

In accordance with 10 CFR 54.21(c)(1)(iii), the Fatigue Monitoring Program will be used to manage the effects of aging due to fatigue on the intended functions of the components associated with fatigue TLAAs for the period of extended operation.

The Fatigue Monitoring Program is an existing program that requires enhancement prior to the period of extended operation.

A.2.1.25 FIRE PROTECTION PROGRAM

The Fire Protection Program is an existing program, described in Appendix F of the FSAR, that detects degradation of components in the scope of license renewal that have fire barrier functions. Periodic visual inspections and functional tests are performed of fire dampers, fire barrier walls, ceilings and floors, fire-rated penetration seals, fire wraps, fire proofing, and fire doors to ensure that functionality and operability are maintained. In addition, the Fire

Protection Program supplements the Fuel Oil Chemistry Program and External Surfaces Monitoring Program through performance monitoring of the diesel-driven fire pump fuel oil supply components and testing and inspection of the halon and carbon dioxide suppression systems, respectively. The Fire Protection Program is a condition monitoring program, comprised of tests and inspections based on National Fire Protection Association (NFPA) recommendations.

A.2.1.26 FIRE WATER PROGRAM

The Fire Water Program (sub-program of the overall Fire Protection Program) is described in Appendix F of the FSAR, and is credited with managing loss of material due to corrosion, erosion, macrofouling, and selective leaching, cracking due to SCC/IGA of susceptible water-based fire 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 Program is a condition monitoring program, comprised of tests and inspections based on NFPA recommendations.

The Fire Water Program is an existing program that requires enhancement prior to the period of extended operation.

A.2.1.27 FLEXIBLE CONNECTION INSPECTION PROGRAM

The Flexible Connection Inspection Program manages degradation, including the effects of the loss of material due to wear and hardening and loss of strength of elastomer components exposed to treated water, dried air, gas, and indoor air environments.

The Flexible Connection Inspection Program is a new plant-specific program that will be implemented via baseline inspection of a sample population followed by opportunistic inspection when components are opened for periodic maintenance, repair, or surveillance activities when surfaces are made available for inspection. These inspections ensure that the existing environmental conditions are not causing material degradation that could result in a loss of component intended function during the period of extended operation. Inspection of a sample population will be conducted within the 10-year period prior to the period of extended operation and will serve as a baseline for future inspections.

A.2.1.28 FLOW-ACCELERATED CORROSION (FAC) PROGRAM

The Flow-Accelerated Corrosion (FAC) Program manages loss of material for steel and gray cast iron components located in the treated water environment of systems that are susceptible to

FAC, also called erosion-corrosion. The FAC Program 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 FAC wear rates. The program is a condition monitoring program that implements the recommendations of NRC Generic Letter 89-08, and follows the guidance and recommendations of EPRI NSAC-202L (Reference A.3-2), to ensure that the integrity of piping systems susceptible to FAC is maintained.

The FAC Program is an existing program that requires enhancement prior to the period of extended operation.

A.2.1.29 FUEL OIL CHEMISTRY PROGRAM

The Fuel Oil Chemistry Program is an existing program that maintains fuel oil quality in order to mitigate degradation of the storage tanks and associated components containing fuel oil that are within the scope of license renewal. The program includes diesel fuel oil testing for emergency diesel generator and diesel-driven fire pump fuel. The Fuel Oil Chemistry Program manages the relevant conditions that could lead to the onset and propagation of loss of material due to corrosion, or cracking due to SCC of susceptible copper alloys, through proper monitoring and control of fuel oil contamination consistent with plant technical specifications and American Society for Testing and 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 of susceptible materials. Exposure to these contaminants is minimized by verifying the quality of new fuel oil before it enters the emergency diesel generator storage tanks and by periodic sampling to ensure that both the emergency diesel generator tanks and fire protection 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 program in mitigating the effects of aging.

A.2.1.30 HEAT EXCHANGERS INSPECTION

The Heat Exchangers Inspection detects and characterizes the surface conditions with respect to fouling of heat exchangers and coolers that are in the scope of the inspection and exposed to indoor air or to water with the chemistry controlled by the BWR Water Chemistry Program or the Closed Cooling Water Chemistry Program. The inspection provides direct evidence as to whether, and to what extent, a reduction of heat transfer due to fouling has occurred on the heat transfer surfaces of heat exchangers and coolers.

The Heat Exchangers 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.

A.2.1.31 HIGH-VOLTAGE PORCELAIN INSULATORS AGING MANAGEMENT PROGRAM

The High-Voltage Porcelain Insulators Aging Management Program is an existing program that manages the build-up of contamination (hard water residue) on the surfaces of the 115-kV high-voltage insulators located in the transformer yard and the 230-kV high voltage insulators located in the Ashe substation. The program provides for periodic cleaning or recoating of insulators and visual inspection of the coating (if present) on the high-voltage station post insulators between the 115-kV backup transformer and circuit breaker E-CB-TRB located in the station transformer yard. Testing for contamination, and cleaning if required, is conducted on the high voltage station post insulators between the 230-kV overhead line running to Columbia and circuit breaker E-CB-TRS, located in the Ashe substation.

The High-Voltage Porcelain Insulators Aging Management Program is a preventive maintenance program consisting of activities to mitigate potential degradation of the insulation function due to hard water deposits. Uncoated insulators located in the transformer yard are inspected and cleaned every two years. Coated insulators are visually inspected for damage every two years and are re-coated every 10 years. The program requires enhancement prior to the period of extended operation to have the insulators located in the Ashe substation tested for contamination, and cleaned if required, every 8 years.

A.2.1.32 INACCESSIBLE POWER CABLES NOT SUBJECT TO 10 CFR 50.49 EQ REQUIREMENTS PROGRAM

The Inaccessible Power Cables Not Subject to 10 CFR 50.49 EQ Requirements Program will manage the aging of in-scope, power cables (≥ 400V) exposed to significant moisture. First tests or first inspection for license renewal will be completed before the period of extended operation. These cables will be tested at least once every 6 years to provide an indication of the condition of the conductor insulation. The specific type of test performed will be determined prior to the initial test, and is to be a proven test for detecting deterioration of the insulation system due to wetting, such as power factor, partial discharge, or polarization index, as described in EPRI TR-103834-P1-2 (Reference A.3-3), or other testing that is state-of-the-art at the time the test is performed. Significant moisture is defined as periodic exposures that last more than a few days (e.g., cable in standing water). Periodic exposures that last less than a few days (e.g., normal rain and drain) are not significant. In addition, inspection for water collection in electrical manholes will be performed based on actual plant experience with water accumulation in the manholes. However, the inspection frequency will

be at least annually. Manhole inspection will also be performed periodically, in response to event-driven occurrences (such as heavy rain or flooding). The inspection will include direct observation that cables are not wetted or submerged, that cables/splices and cable support structures are intact, and sump pump systems and associated alarms operate properly. In addition, sump pumps will be inspected and operation verified prior to any known or predicted heavy rain or flooding events which could require the sump pump to operate.

A.2.1.33 INSERVICE INSPECTION (ISI) PROGRAM

The Inservice Inspection (ISI) Program is an existing condition monitoring program that manages cracking due to SCC/IGA and flaw growth 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 also manages loss of material due to corrosion for reactor vessel internals components and reduction of fracture toughness due to thermal embrittlement of cast austenitic stainless steel pump casings and valve bodies.

The Inservice Inspection (ISI) Program details the requirements for the examination, testing, repair, and replacement of components specified in ASME Section XI for Class 1, 2, or 3 components. The Inservice Inspection (ISI) Program complies with the ASME Code requirements.

The program scope has been augmented to include additional requirements, and components, beyond the ASME requirements. Examples include the augmentation of ISI to expand reactor vessel feedwater nozzle examinations, examinations of high energy line piping systems that penetrate containment, examinations per Generic Letter 88-01, and examinations of shroud support plate access hole covers per BWRVIP guidance. Such augmentation is consistent with the ISI program description in NUREG-1801, Section XI.M1.

A.2.1.34 INSERVICE INSPECTION (ISI) PROGRAM – IWE

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 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, Subsection IWE.

The inservice examinations conducted throughout the service life of Columbia will comply with the requirements of the ASME Section XI Edition and Addenda incorporated by reference in 10 CFR 50.55a(b) twelve months prior to the start of the inspection interval, subject to prior

approval of the edition and addenda by 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.

A.2.1.35 INSERVICE INSPECTION (ISI) PROGRAM – IWF

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 performs visual examination of supports 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 and those other than piping supports (Class 1, 2, 3, and MC)). 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 the ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWF.

The inservice examinations conducted throughout the service life of Columbia will comply with the requirements of the ASME Section XI Edition and Addenda incorporated by reference in 10 CFR 50.55a(b) twelve months prior to the start of the inspection interval, subject to prior approval of the edition and addenda by 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.

A.2.1.36 LUBRICATING OIL ANALYSIS PROGRAM

The Lubricating Oil Analysis Program manages loss of material due to corrosion or selective leaching of susceptible materials and reduction of heat transfer due to fouling for plant components that are within the scope of license renewal and exposed to a lubricating oil environment. The Lubricating Oil Analysis Program is a mitigation program.

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

The Lubricating Oil Analysis Program is an existing program that requires enhancement prior to the period of extended operation.

A.2.1.37 LUBRICATING OIL 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 due to corrosion or selective leaching has occurred. The inspection also determines whether 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.

A.2.1.38 MASONRY WALL INSPECTION

The Masonry Wall Inspection 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 Inspection is implemented as part of the Structures Monitoring Program. The Masonry Wall Inspection performs visual inspection of external surfaces of masonry walls.

The Masonry Wall Inspection is an existing program that requires enhancement prior to the period of extended operation.

A.2.1.39 MATERIAL HANDLING SYSTEM INSPECTION PROGRAM

The Material Handling System Inspection Program manages loss of material for cranes (including bridge, trolley, rails, and girders), monorails, and hoists within the scope of license renewal. The Material Handling System 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.

A.2.1.40 METAL-ENCLOSED BUS PROGRAM

The Metal-Enclosed Bus Program is an inspection program that detects degradation of metal-enclosed bus within the scope of license renewal. The program provides for the visual inspection of interior sections of bus, and an inspection of the elastomeric seals at the joints of the duct sections. The program also makes provision for thermographic inspection of bus bolted connections.

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

program will be performed once every 10 years, with the initial inspections to be performed prior to the period of extended operation. The visual inspection portion of the program will also be performed once every 10 years, with the first inspections to be performed prior to the period of extended operation.

A.2.1.41 MONITORING AND COLLECTION SYSTEMS INSPECTION PROGRAM

The Monitoring and Collection Systems Inspection Program manages the effects of the loss of material due to corrosion or erosion for the internal surfaces of subject mechanical components that are exposed to equipment or area drainage water and other potential contaminants and fluids. The inspection also manages cracking due to SCC of susceptible materials.

The Monitoring and Collection Systems Inspection Program is a program that will be implemented via baseline inspection of a sample population followed by opportunistic inspection when components are opened for periodic maintenance, repair, or surveillance activities when surfaces are made available for inspection. These inspections ensure that the existing environmental conditions are not causing material degradation that could result in a loss of component intended function during the period of extended operation. Inspection of a sample population will be conducted within the 10-year period prior to the period of extended operation and will serve as a baseline for future inspections.

A.2.1.42 OPEN-CYCLE COOLING WATER PROGRAM

The Open-Cycle Cooling Water Program manages loss of material due to corrosion and erosion for components located in the Standby Service Water and Plant Service Water systems, and for components connected to or serviced by those systems. The program manages fouling due to particulates (e.g., corrosion products) and biological material (micro- or macro-organisms) resulting in reduction in heat transfer for heat exchangers (including condensers, coolers, cooling coils, and evaporators) within the scope of the program. The Open-Cycle Cooling Water Program also manages loss of material for components associated with the feed-and-bleed mode for emergency makeup water to the spray pond.

The Open-Cycle Cooling Water Program consists of inspections, surveillances, and testing to detect the presence, and assess the extent of fouling and loss of material. The inspection activities are combined with chemical treatments and cleaning activities to minimize the effects of aging. The program is a combination condition monitoring and mitigation program that implements the recommendations of NRC Generic Letter 89-13 for safety-related equipment in the scope of the program. The scope of the program also includes non-safety related components containing either service water or spray pond makeup water.

The Open-Cycle Cooling Water Program is an existing program that requires enhancement prior to the period of extended operation.

A.2.1.43 POTABLE WATER MONITORING PROGRAM

The Potable Water Monitoring Program is a mitigation program that, by means of chemical water treatment, manages loss of material due to corrosion and erosion for components that contain potable water.

The Potable Water Monitoring Program is an existing program that requires enhancement prior to the period of extended operation. At least one inspection will be conducted within the 10-year period prior to the period of extended operation.

A.2.1.44 PREVENTIVE MAINTENANCE – RCIC TURBINE CASING

Preventive Maintenance – RCIC Turbine Casing is an existing program that manages loss of material due to corrosion for the reactor core isolation cooling (RCIC) pump turbine casing and associated piping components downstream from the steam admission valve. These components are exposed to steam during RCIC system operation and testing, but are empty during normal plant operating conditions. Preventive Maintenance – RCIC Turbine Casing is a condition monitoring program comprised of periodic inspection and surveillance activities to detect aging and age-related degradation.

A.2.1.45 REACTOR HEAD CLOSURE STUDS PROGRAM

The Reactor Head Closure Studs Program is an existing program that manages cracking due to SCC and loss of material due to corrosion for the reactor head closure stud assemblies (studs, nuts, washers, and bushings). The Reactor Head Closure Studs Program examines reactor vessel stud assemblies in accordance with the examination and inspection requirements specified in the ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWB (edition and addenda described in the Inservice Inspection (ISI) Program), Table IWB 2500-1. The Reactor Head Closure Studs Program includes preventive measures in accordance with Regulatory Guide 1.65 to mitigate cracking.

The Reactor Head Closure Studs Program credits portions of the Inservice (ISI) Inspection Program.

A.2.1.46 REACTOR VESSEL SURVEILLANCE PROGRAM

The Reactor Vessel Surveillance Program is an existing condition monitoring program that manages reduction of fracture toughness due to radiation embrittlement for the low alloy steel

reactor vessel shell and welds in the beltline region. The Reactor Vessel Surveillance Program incorporates the BWRVIP Integrated Surveillance Program (ISP), as described in reports BWRVIP-86-A and BWRVIP-116.

Energy Northwest follows the requirements of the BWRVIP ISP and applies the ISP data to Columbia. The NRC has approved the use of the BWRVIP ISP in place of a unique plant program for Columbia.

The provisions of 10 CFR 50 Appendix G require Columbia to operate within the currently licensed pressure-temperature (P-T) limit curves, and to update these curves as necessary. The P-T limit curves, as contained in plant technical specifications, will be updated as necessary through the period of extended operation as part of the Reactor Vessel Surveillance Program. Reactor vessel P-T limits will thus be managed for the period of extended operation.

A.2.1.47 SELECTIVE LEACHING INSPECTION

The Selective Leaching Inspection detects and characterizes the conditions on internal and external surfaces of subject components exposed to raw water, treated water, fuel oil, soil, and moist air (including condensation) environments. The inspection provides direct evidence through a combination of visual examination and hardness testing, or NRC-approved alternative, as to whether, and to what extent, a loss of material due to selective leaching has occurred.

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 no earlier than 5 years prior to the period of extended operation.

A.2.1.48 SERVICE AIR SYSTEM INSPECTION PROGRAM

The Service Air System Inspection Program manages the effects of the loss of material due to corrosion of steel piping and valve bodies exposed to an “air (internal)” (i.e., compressed air) environment within the license renewal boundary of the Service Air System.

The Service Air System Inspection Program is a new plant-specific program that will be implemented via baseline inspection of a sample population followed by opportunistic inspection when components are opened for periodic maintenance, repair, or surveillance activities when surfaces are made available for inspection. These inspections ensure that the existing environmental conditions are not causing material degradation that could result in a loss of component intended function during the period of extended operation. Inspection of a sample population will be conducted within the 10-year period prior to the period of extended operation and will serve as a baseline for future inspections.

A.2.1.49 SMALL BORE CLASS 1 PIPING PROGRAM

The Small Bore Class 1 Piping Program will detect and characterize cracking of small bore Class 1 piping components that are exposed to reactor coolant. This periodic program will provide physical evidence as to whether, and to what extent, cracking due to SCC or to thermal or mechanical loading has occurred in small bore Class 1 piping components. The Small Bore Class 1 Piping Program will be a condition monitoring program with no actions to prevent or mitigate aging effect. The program will include visual and volumetric inspection of a representative sample of small bore Class 1 piping, including butt welds and socket welds.

The Small Bore Class 1 Piping Program is a new program that will be implemented prior to the period of extended operation. Inspection activities will start during the fourth 10-year inservice inspection interval and continue through the period of extended operation. The Small Bore Class 1 Piping Inspection will credit portions of the Inservice Inspection (ISI) Program. The Small Bore Class 1 Piping Inspection will verify the effectiveness of the BWR Water Chemistry Program in mitigating cracking of small bore piping and piping components.

A.2.1.50 STRUCTURES MONITORING PROGRAM

The Structures Monitoring Program 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. The Structures Monitoring Program encompasses and implements the Water Control Structures Inspection and the Masonry Wall Inspection. 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.

The Structures Monitoring Program is an existing program that requires enhancement prior to the period of extended operation.

A.2.1.51 SUPPLEMENTAL PIPING/TANK INSPECTION

The Supplemental Piping/Tank Inspection detects and characterizes the material condition of steel, gray cast iron, and stainless steel components exposed to moist air environments. The inspection provides direct evidence as to whether, and to what extent, a loss of material due to corrosion has occurred.

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.

A.2.1.52 THERMAL AGING AND NEUTRON EMBRITTLEMENT OF CAST AUSTENITIC STAINLESS STEEL (CASS) PROGRAM

The Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steel (CASS) Program will manage reduction of fracture toughness due to thermal aging and neutron irradiation embrittlement of CASS reactor vessel internals.

The program includes: (a) identification of susceptible components determined to be limiting from the standpoint of thermal aging or neutron irradiation embrittlement (neutron fluence), (b) a component-specific evaluation to determine each identified component's susceptibility to reduction of fracture toughness, and (c) a supplemental examination of any component not eliminated by the component-specific evaluation.

The program credits portions of the Inservice Inspection (ISI) Program and the BWR Vessel Internals Program.

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.

A.2.1.53 WATER CONTROL STRUCTURES INSPECTION

The Water Control Structures Inspection, implemented as part of the Structures Monitoring Program, consists of inspection activities to detect aging and age-related degradation. The Water Control Structures Inspection ensures the structural integrity and operational adequacy of the spray ponds, standby service water pump houses, circulating water pump house (including circulating water basin), makeup water pump house, cooling tower basins, and those structural components within the structures.

The Water Control Structures Inspection is an existing program that requires enhancement prior to the period of extended operation.

A.2.1.54 BORON CARBIDE MONITORING PROGRAM

The Boron Carbide Monitoring Program detects degradation of the Boron Carbide (B_4C) neutron absorbers in the spent fuel storage racks by monitoring spent fuel racks for potential off-gassing, by in situ testing of the spent fuel racks, or by inspecting the B_4C coupons. From

the monitoring data, the stability and integrity of Boron Carbide in the storage cells are assessed. Periodic monitoring of B_4C coupons permits early determination of aging degradation, but may be discontinued based on in situ testing results.

A.2.1.55 SERVICE LEVEL 1 PROTECTIVE COATINGS PROGRAM

The Service Level 1 Protective Coating Program monitors the performance of Service Level 1 coatings inside containment through periodic coating examinations, condition assessments, and remedial actions, including repair or testing. The program establishes roles, responsibilities, controls and deliverables for the Service Level 1 Protective Coatings Program. This program also ensures the Design Basis Accident (DBA) analysis limits with regard to coating will not be exceeded for the suction strainers.

A.2.2 EVALUATION OF TIME-LIMITED AGING ANALYSES

In accordance with 10 CFR 54.21(c), an application for a renewed operating license requires an evaluation of TLAAs for the period of extended operation. The following TLAAs have been identified and evaluated to meet this requirement.

A.2.2.1 REACTOR VESSEL NEUTRON EMBRITTLEMENT

Neutron embrittlement is the change in mechanical properties of reactor vessel materials resulting from exposure to fast neutron flux ($E > 1.0$ MeV) in the beltline region of the reactor core. The most pronounced material change is a reduction in fracture toughness. As fracture toughness decreases with cumulative fast neutron exposure, the material's resistance to crack propagation decreases. Fracture toughness is also dependent on temperature. The reference temperature for nil-ductility transition (RT_{NDT}) is the temperature above which the material behaves in a ductile manner and below which the material behaves in a brittle manner. As fluence increases, RT_{NDT} increases, and higher temperatures are required for the material to continue to act in a ductile manner.

Requirements associated with fracture toughness, pressure-temperature limits, and material surveillance programs for the reactor coolant pressure boundary are contained in Appendices G and H of 10 CFR 50.

The analyses associated with evaluation of the effect of neutron embrittlement on the reactor pressure vessel for 40 years are TLAAs. Neutron fluence, upper shelf energy, adjusted reference temperature (ART), and vessel P-T limits are time dependent parameters associated with fracture toughness (embrittlement) of reactor vessel materials.

A.2.2.1.1 Neutron Fluence

EFPY Projection

To evaluate the effects of radiation on reactor pressure vessel material embrittlement, the results of analyses were projected to determine neutron fluence out to 54 effective full power years (EFPY). Using actual reactor core power histories through 2007 and conservative estimates of future core designs, extended operation to 60 years was determined to be bounded by 54 EFPY.

Fluence Projection

Analyzed fluence values at 51.6 EFPY of reactor operation are addressed in FSAR Section 4.3.2.8 and FSAR Table 4.3-1. These fluence analyses are based on the original licensed thermal power of 3323 mega-watt thermal (MWt) through fuel cycle 10, and the currently licensed thermal power uprated to 3486 MWt from cycle 11 through the end of operation. These fluence analyses use NRC-approved methodology based on the guidance of Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." The fluence analyses were projected to 54 EFPY for the extended operating period of 60 years.

Beltline Evaluation

For the extended operating period, all ferritic materials for vessel beltline shells, welds, nozzles and the associated nozzle to vessel welds, and assembly components are required to be evaluated for neutron irradiation embrittlement if high energy neutron fluence is greater than a threshold value of 1E+17 n/cm^2 (E >1 MeV) at the end of the 60 years. The only vessel assembly items, other than the shells and welds of the beltline region that would experience neutron fluence greater than 1E+17 n/cm^2 during the period of extended operation are instrumentation nozzle N12 and residual heat removal/low pressure coolant injection (RHR/LPCI) nozzle N6 (and the associated nozzle-to-vessel welds).

Instrumentation nozzle N12 has a thickness less than 2.5 inches and was not originally evaluated for fracture toughness per ASME Code Appendix G, Section G2223. Nozzle N12 is not limiting for P-T curves as discussed in Section A.2.2.1.4; however, as nozzle N12 was evaluated for impact on the P-T curves it meets the definition of a beltline component per 10 CFR 50, Appendix G. The associated nozzle-to-vessel weld is an austenitic weld and, therefore, is not subject to the fracture toughness requirements of 10 CFR 50, Appendix G.

Nozzle N6 is included in the evaluation for USE in Section A.2.2.1.2. Nozzle N6 is evaluated for ART in Section A.2.2.1.3 below. Nozzle N6 is not the limiting material for the vessel. However, as nozzle N6 was evaluated for ART it meets the definition of a beltline component

per 10 CFR 50, Appendix G. The associated nozzle-to-vessel weld is a ferritic weld and, therefore, is subject to the fracture toughness requirements of 10 CFR 50, Appendix G. The nozzle-to-vessel weld for nozzle N6 is also included in the evaluation for USE in Section A.2.2.1.2 and is evaluated for ART in Section A.2.2.1.3.

The beltline definition for the period of extended operation includes the lower shell (Course #1 / Ring #21), lower-intermediate shell (Course #2 / Ring #22), associated vertical (longitudinal) welds, the girth (circumferential) weld that connects the lower and lower-intermediate shells, and nozzles N6 (and its associated nozzle-to-vessel weld) and nozzle N12.

Disposition

Neutron fluence has been projected to the end of the period of extended operation.

A.2.2.1.2 Upper Shelf Energy Evaluation

Appendix G of 10 CFR 50 requires the upper shelf energy (USE) of the vessel beltline materials to remain above 50 ft-lb at all times during plant operation, including the effects of neutron radiation. If USE cannot be shown to remain above this limit, then an equivalent margin analysis (EMA) must be performed to show that the margins of safety against fracture are equivalent to those required by Appendix G of Section XI of the ASME Code.

The initial (unirradiated) USE is not known for all the Columbia vessel plates and welds. For those plates and welds for which the initial USE is known, USE was projected using Regulatory Guide 1.99, Revision 2 methods. For the vessel plates and welds for which the initial USE is not known, USE equivalent margin analyses were performed using the Boiling Water Reactor Owners Group (BWROG) equivalent margin analysis (EMA) methodology. Results from the testing and analysis of surveillance materials were used in the EMA analyses.

All of the projected USE values for the vessel beltline plates, nozzle forgings, and welds for which the initial USE is known remain above 50 ft-lbs through the end of the period of extended operation (54 EFPY). For the vessel beltline plates and welds, for which the initial USE is not known, the maximum decrease in USE was found to be less than the assumed decrease in the associated generic equivalent margin analyses. The maximum predicted decreases in USE for 54 EFPY for these beltline plates and welds are bounded by the generic equivalent margin analyses. Therefore, the projected USE for the vessel beltline plates and welds is acceptable for the period of extended operation.

Energy Northwest agrees that all beltline materials, including the N12 instrumentation nozzles, must be considered when the licensee develops pressure-temperature limits for Columbia in accordance with 10 CFR Part 50, Appendix G and ASME Code, Section XI, Appendix G.

Energy Northwest will continue to develop future pressure-temperature limit curves considering all beltline plates, welds, and nozzles.

<u>Disposition</u>

Upper shelf energy TLAAs have been projected to the end of the period of extended operation for all reactor vessel beltline materials. Additionally, a specific 54 EFPY equivalent margins analysis will be performed for the N12 nozzle forgings prior to the period of extended operation.

A.2.2.1.3 Adjusted Reference Temperature Analysis

In addition to USE, the other key parameter that characterizes the fracture toughness of a material is the RT_{NDT}. This reference temperature changes as a function of exposure to neutron radiation resulting in an adjusted reference temperature, ART.

The initial RT_{NDT} is the reference temperature for the unirradiated material. The change due to neutron radiation is referred to as ΔRT_{NDT}. The ART is calculated by adding the initial RT_{NDT}, the ΔRT_{NDT}, and a margin to account for uncertainties as prescribed in Regulatory Guide 1.99, Revision 2.

The ART evaluations of record for the vessel beltline plates, nozzle forgings, and welds for the currently licensed period (33.1 EFPY) include power uprate conditions. Based on projected fluence values, the methodology in Regulatory Guide 1.99 was used to project the ART for 54 EFPY. The ART values projected to 54 EFPY are used to develop P-T limit curves. Projected ART values are well below the 200°F end of life ART suggested in Section 3 of Regulatory Guide 1.99 and are, thus, acceptable for the period of extended operation.

<u>Disposition</u>

Reactor vessel adjusted reference temperature TLAAs have been projected to the end of the period of extended operation.

A.2.2.1.4 Pressure-Temperature Limits

To ensure 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.

The Columbia P-T limit curves were revised in 2005 to include the effects of power uprate to 3486 MWt. The P-T limits are valid for 33.1 EFPY through the end of the currently licensed period. The curves were reviewed in 2009 to assure that the N12 instrumentation nozzle did not affect the existing curves. P-T limits for the period of extended operation will be calculated using the most accurate fluence projections available at the time of the recalculation. The projections may be adjusted if there are changes in core design or if additional surveillance capsule results show the need for an adjustment. The projected ART for the period of extended operation above gives confidence that future P-T curves will provide adequate operating margin.

Energy Northwest will continue to develop pressure-temperature limits in accordance with the Title 10 of the Code of Federal Regulations Part 50, Appendix G (10 CFR Part 50, Appendix G) and ASME Code, Section XI, Appendix G, considering all beltline plates, welds, and nozzles.

License amendment requests to revise the P-T limits will be submitted to the NRC for approval, when necessary to comply with 10 CFR 50 Appendix G, as part of the Reactor Vessel Surveillance Program.

Disposition

The TLAA for P-T limits will be adequately managed for the period of extended operation as part of the Reactor Vessel Surveillance Program.

A.2.2.1.5 Reactor Vessel Circumferential Weld Inspection Relief

BWRVIP-74-A, “BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines for License Renewal,” reiterated the recommendation of BWRVIP-05, “BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations,” that vessel circumferential welds could be exempted from examination. The NRC safety evaluation report (SER) for BWRVIP-74 agreed, but required that plants apply for this relief request individually. The relief request is required to demonstrate that at the expiration of the current license, the circumferential welds will satisfy the limiting conditional failure probability in the (BWRVIP-05) evaluation. Energy Northwest requested and received permanent relief from vessel shell circumferential (girth) weld volumetric examinations through 33.1 EFPY.

The reactor pressure vessel circumferential weld parameters at 54 EFPY have been projected to remain within the bounding (64 EFPY) vessel parameters from the BWRVIP-05 SER. As such, the conditional probability of failure for circumferential welds remains below the limits contained in the SER for BWRVIP-05.

Disposition

The TLAA for reactor vessel circumferential weld examination relief has been projected to the end of the period of extended operation.

A.2.2.1.6 Reactor Vessel Axial Weld Failure Probability

The NRC SER for BWRVIP-74-A evaluated the failure frequency of axially oriented welds in BWR reactor vessels, and determined failure frequency acceptance criteria for 40 years of reactor operation. Applicants for license renewal are required to evaluate axially oriented vessel welds to show that their failure frequency remains below the acceptance criteria in the SER 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 SER.

The Columbia limiting axial weld mean RT_{NDT} at 54 EFPY is projected to remain well below the RT_{NDT} from the SER for BWRVIP-74, thus the Columbia axial weld failure frequency meets the acceptable criteria.

Disposition

The TLAA for the reactor vessel axial weld failure probability has been projected to the end of the period of extended operation.

A.2.2.2 METAL FATIGUE

Fatigue evaluations for mechanical components are identified as TLAAs; therefore, the effects of fatigue must be addressed for license renewal. Fatigue is an age-related degradation mechanism caused by cyclic duty on a component by either mechanical or thermal loads.

The ASME Boiler and Pressure Vessel Code requires evaluation of transient thermal and mechanical load cycles for Class 1 components. Cumulative usage factors for Class 1 components are calculated based on normal and upset design transient definitions. The design transients used to generate cumulative usage factors for Class 1 components are contained in FSAR Section 3.9.1.1. Energy Northwest is required to monitor design transients listed in FSAR Table 3.9-1 to ensure that plant components are maintained within the design limits.

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.

The reactor coolant environmental effects of fatigue on plant components were also evaluated.

The design cycles for Columbia are summarized in FSAR Section 3.9 and FSAR Table 3.9-1. Energy Northwest counts all fatigue significant cycles, not only for the design transients listed in FSAR Table 3.9-1 but also for the analysis of other plant components. The events listed in FSAR Table 3.9-1 have been evaluated and in some cases regrouped for easier counting. Faulted conditions listed in the FSAR are not used in the fatigue analyses and are not counted. Additional transients determined to be fatigue significant after the original design have been added to the counting procedure, while FSAR Table 3.9-1 lists the original design cycles. The projected number of occurrences of design transients to 60 years determined that some analyzed numbers of transients may be exceeded. These projections were done using linear extrapolation from the beginning of plant life. Recent operating experience suggests lower projections and as additional operating data is accumulated, subsequent projections will refine the number of cycles expected in 60 years. Energy Northwest manages fatigue using the Fatigue Monitoring Program to track transient cycles and require corrective action before any analyzed number of cycles is reached.

A.2.2.2.1 Reactor Pressure Vessel Fatigue Analyses

The reactor vessel assembly consists of the reactor pressure vessel (RPV), the vessel support skirt, the shroud support, nozzles, penetrations, stub tubes, head closure flanges, head closure studs, refueling bellows support, and stabilizer brackets.

Design cumulative usage factors (CUFs) for the limiting RPV assembly locations are contained in design reports and were calculated based on the design transients. Energy Northwest manages fatigue for the RPV assembly components using the Fatigue Monitoring Program to track transient cycles and requires corrective action before any analyzed number of cycles is reached.

<u>Disposition</u>

The effects of aging on the intended functions of the RPV will be adequately managed for the period of extended operation by the Fatigue Monitoring Program.

A.2.2.2.2 Reactor Pressure Vessel Internals

Fatigue analyses of the overall RPV internals (including the jet pump assemblies) were performed pre-startup as part of the plant design. Component specific fatigue analyses of the jet pumps were performed more recently to bound actual plant operation. Each of these analyses is discussed below.

Reactor Vessel Internals Fatigue Analyses

The RPV internals are described in terms of two assemblies: core support structures and reactor internals. Core support structures include the shroud, shroud support (included as part of the reactor vessel for fatigue), core plate with core plate hold-down bolts, top guide, fuel supports, and control rod guide tubes. Reactor internals include the jet pump assemblies, jet pump instrumentation, feedwater spargers, vessel head spray nozzle, differential pressure line, incore flux monitor guide tubes, surveillance sample holders, core spray line (in-vessel) and spargers, incore instrument housings, low pressure coolant injection coupling, steam dryer, shroud head and steam separator assembly, guide rods, and control rod drive thermal sleeves.

The normal, test, and upset service load cycles used for the design and fatigue analysis for the core support structures and reactor internals are shown in FSAR Table 3.9-1. Calculation of CUFs for the reactor internals was performed as part of a NSSS design evaluation.

Review of the RPV internals in association with power uprate determined that stresses on the vessel internals remained well below all limits. No recalculation of cumulative usage factors was determined to be required. Energy Northwest manages fatigue using the Fatigue Monitoring Program to track transient cycles and require corrective action before any analyzed number of cycles is reached.

Disposition

The effects of aging on the intended functions of the RPV internals will be adequately managed for the period of extended operation by the Fatigue Monitoring Program.

Jet Pump Fatigue Analyses

In August 2000, Columbia operated for a period of time with the recirculation pumps in an unbalanced mode (pump speeds different by more than 50 percent). The effect of that flow imbalance on the jet pumps was an additional accumulation of fatigue usage.

As a result of inspections during the Spring 2001 outage (R-15), a fatigue analysis of the jet pumps was performed and cumulative usage factors were determined.

Jet pump clamps were installed during the 2005 outage (R-17) to minimize flow induced vibration. These clamps greatly reduced the future potential for riser brace fatigue.

As a result of evaluations after the 2007 outage the usage factors were extended to 60 years. The maximum CUF of the jet pump risers for 60 years of operation is projected to remain below the fatigue limit. Energy Northwest manages fatigue using the Fatigue Monitoring Program to track transient cycles and require corrective action before any analyzed number of cycles is reached. The Fatigue Monitoring Program credits the BWR Vessel Internals Program

to monitor the jet pump gaps. Together, these actions effectively manage the fatigue of the jet pumps through the period of extended operation.

Disposition

The effects of aging on the intended functions of the jet pumps will be adequately managed for the period of extended operation by the Fatigue Monitoring Program.

A.2.2.2.3 Reactor Coolant Pressure Boundary Piping and Piping 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. Fatigue analyses of Class 1 piping are based on the transients found in the Columbia piping specifications that are in turn based on the design transients listed in FSAR Section 3.9.

Potential high energy line break (HELB) intermediate locations can be eliminated based on CUFs of less than 0.1 if other stress criteria are also met. The usage factors, as calculated in the design fatigue analyses, account for the design transients assumed for the original 40-year life of the plant. Therefore, the determination of CUFs used in the selection of postulated high energy line intermediate break locations are TLAAs. The Fatigue Monitoring Program will identify when the transients for piping systems are approaching their analyzed number of cycles. Prior to any transient exceeding its analyzed number of cycles for a piping system, the associated analyses will be reviewed to determine whether any additional locations need to be designated as postulated HELB locations.

All Class 1 piping was reviewed for the power uprate. The evaluation determined that there was adequate margin in each system to accommodate the power uprate. Design fatigue usage for 40 years of operation and projected fatigue usage for the period of extended operation are established for the limiting reactor coolant pressure boundary components.

A review of documentation found several fatigue analyses for Class 1 valve stress reports found fatigue analyses that were TLAAs. The fatigue usage for those valves is based on transients that are tracked by the Fatigue Monitoring Program.

Metal fatigue for all Class 1 reactor coolant pressure boundary piping and in-line components is managed by the Fatigue Monitoring Program. The Fatigue Monitoring Program will identify when the transients for piping systems are approaching their analyzed numbers of cycles. Prior to any transient exceeding its analyzed number of cycles for a piping system, the design calculations for that system will be reviewed and appropriate actions will be taken.

Disposition

The effects of aging on the intended functions of the reactor coolant pressure boundary piping and components will be adequately managed for the period of extended operation by the Fatigue Monitoring Program.

A.2.2.3 NON-CLASS 1 COMPONENT FATIGUE ANALYSES

The non-Class 1 mechanical components susceptible to fatigue fit into one of two major categories: (1) piping and in-line components (piping, valves, tubing, traps, thermowells, etc.) or (2) non-piping components (vessels, heat exchangers, tanks, pumps, etc.).

Non-Class 1 components that are Quality Group B or C are designed and constructed as ASME Section III Code Class 2 and 3, respectively. The design of ASME Class 2 and 3 piping systems incorporates a stress range reduction factor for determining acceptability of piping design with respect to thermal stresses. Non-Class 1 components designated as Quality Class D are designed to ANSI B31.1, which also incorporates stress range reduction factors based upon the number of thermal cycles. In general, a stress range reduction factor of 1.0 in the stress analyses applies for up to 7,000 thermal cycles. The allowable stress range is reduced by the stress range reduction factor if the number of thermal cycles exceeds 7,000. If fewer than 7,000 cycles are expected through the period of extended operation, then the fatigue analysis (stress range reduction factor) of record will remain valid through the period of extended operation.

Because 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, no fatigue evaluation is required. Therefore, there are no fatigue TLAAs for these components.

The fatigue evaluation of non-Class 1 piping and in-line components evaluated the associated operating temperature against the threshold temperature value for fatigue of the material. If the threshold temperature value was exceeded, then the number of transient cycles for the piping or in-line component was projected. In each case, the number of projected cycles for 60 years was found to be less than 7,000 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.

Disposition

The TLAA for non-Class 1 component fatigue analyses remains valid for the period of extended operation.

A.2.2.4 EFFECTS OF REACTOR COOLANT ENVIRONMENT ON FATIGUE LIFE OF COMPONENTS AND PIPING

Applicants for license renewal are required to address the reactor coolant environmental effects on fatigue of plant components. The minimum set of components for a BWR of Columbia's vintage is derived from NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," as follows:

1. Reactor vessel shell and lower head
2. Reactor vessel feedwater nozzle
3. Reactor recirculation piping (including inlet and outlet nozzles)
4. Core spray line reactor vessel nozzle and associated Class 1 piping
5. Residual heat removal return line Class 1 piping
6. Feedwater line Class 1 piping

Energy Northwest has analyzed these locations for the effects of the reactor coolant environment on fatigue in support of license renewal. Energy Northwest has also analyzed other limiting components beyond those locations indentified in NUREG/CR-6260 for the effects of the reactor coolant environment. Original fatigue usage calculations were reviewed, and the transient groupings and load pairs used in those analyses were carried over to the environmentally-assisted fatigue analyses, with revised non-environmentally assisted usage factors determined.

An effective fatigue life adjustment factor, F_{en}, that considers a time weighted average of operation with normal water chemistry and hydrogen water chemistry over 60 years of operation, was determined for each load pair analyzed for the components. The fatigue life adjustment factors were applied to the revised component load pair usage factors, and the environmentally-adjusted usage factors were summed to obtain environmentally-adjusted CUFs to verify acceptability of the components for the period of extended operation.

Using fatigue data projected by the Fatigue Monitoring Program and the methodology summarized above, the limiting locations were evaluated. None of the locations evaluated have an environmentally adjusted CUF of greater than 1.0 during the period of extended operation.

For the period of extended operation, on an ongoing basis, ensure that all the limiting locations in Class 1 components and Class 1 systems have been evaluated for the effect of reactor water environment.

The aging effect of fatigue, including consideration of the environmental effects, will be adequately managed for the period of extended operation using the Fatigue Monitoring Program.

Disposition

The effects of environmentally-assisted fatigue on the intended functions of the NUREG/CR-6260 and other limiting locations will be adequately managed for the period of extended operation using the Fatigue Monitoring Program.

A.2.2.5 ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT

Environmental qualification analyses for electrical equipment are identified as TLAAs. 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. The EQ Program implements the requirements of 10 CFR 50.49 and will be used to manage the effects of aging on the intended functions of the components associated with environmental qualification TLAAs for the period of extended operation.

Disposition

The effects of aging on the intended functions of the environmentally qualified components will be adequately managed for the period of extended operation by the EQ Program.

A.2.2.6 FATIGUE OF PRIMARY CONTAINMENT, ATTACHED PIPING, AND COMPONENTS

The Primary Containment and attached piping and components susceptible to fatigue resulting from the effects of plant transients are evaluated below.

A.2.2.6.1 Primary Containment

The cycles used in the fatigue evaluation of the containment components are provided in FSAR Table 3A.4.1-3. No operating basis earthquakes have been experienced by Columbia through 2007, and the containment analysis for five operating basis earthquakes remains valid for 60 years of plant operation. The safe shutdown earthquake and post-loss of coolant accident (LOCA) chugging are once in a lifetime events and are not projected to occur during the extended period of operation. Safety relief valve actuations have been projected through 60 years of operation based on the number of actual events through 2007. The fatigue analyses performed using these events will remain valid for the period of extended operation.

As the cycles on which the containment fatigue analysis is based will not be exceeded for 60 years of operation, these analyses will remain valid for the period of extended operation.

Disposition

The TLAA associated with fatigue of the containment remains valid for the period of extended operation.

A.2.2.6.2 ASME Class MC Components

Class MC components include the primary containment vessel shell, large openings (equipment hatch, personnel hatches, and access hatch), penetrations (all except the large openings), and attachments (pipe supports in the wetwell, welding pads in the drywell, supports for the stabilizer truss, seal and shear lugs at the drywell floor, supports for the downcomer bracing system, pipe whip supports, radial beam supports, cap truss supports, catwalks, monorail, and platforms). The Class MC components were analyzed for fatigue using the transients listed in FSAR Table 3A.4.1-3. As these cycles will not be exceeded for 60 years of operation, the Class MC component fatigue analysis will remain valid for the period of extended operation.

A specific fatigue analysis was performed for the main steam penetrations using the transients listed in FSAR Table 3A.4.1-3. This analysis will remain valid for the period of extended operation as these cycles will not be exceeded for 60 years of operation.

The effects of power uprate on the containment system response were reviewed and determined to be negligible. The containment peak pressure values remain virtually unaffected by the power uprate and extended load line limit. The LOCA containment dynamic loads are not affected by power uprate, and safety relief valve containment loads will remain below their design allowables. (See FSAR Section 3A.)

All events, including safety relief valve actuations, for 60 years of operation are projected to remain below the containment cyclic basis from FSAR Table 3A.4.1-3. Consequently, the analysis of the Class MC containment components remains valid for the period of extended operation.

Disposition

The TLAAs for fatigue of the ASME Class MC components remain valid through the end of the period of extended operation.

A.2.2.6.3 Downcomers

Although not an ASME Code requirement, a fatigue evaluation of the downcomers was performed. The fatigue evaluation of the downcomer lines in the wetwell air volume was

based on the number of cycles presented in FSAR Table 3A.4.1-3. The maximum fatigue usage factor for the downcomers is provided in FSAR Table 3A.4.2-4 and FSAR Table 3A.4.2-5.

All events, including safety relief valve actuations, for 60 years of operation are projected to remain below the containment cyclic basis from FSAR Table 3A.4.1-3. Consequently, the analysis of the downcomers remains valid for the period of extended operation.

Disposition

The TLAA for fatigue of the downcomers remains valid through the end of the period of extended operation.

A.2.2.6.4 Safety Relief Valve Discharge Piping

Although not an ASME Code requirement, a fatigue evaluation of the safety relief valve (SRV) discharge piping was performed. The fatigue evaluation used the number of cycles as presented in FSAR Table 3A.4.1-3. The maximum fatigue usage factor for all 18 SRV discharge lines in the wetwell air volume is below the ASME allowable limits per FSAR Section 3A.4.2.4.6.

The SRV actuations for 60 years of operation are projected to remain below the containment cyclic basis from FSAR Table 3A.4.1-3. Consequently, the analysis of the SRV discharge piping remains valid for the period of extended operation.

Disposition

The TLAA for fatigue of the SRV discharge piping remains valid through the end of the period of extended operation.

A.2.2.6.5 Diaphragm Floor Seal

The diaphragm floor seal is located at the inside surface of the primary containment vessel periphery. It provides a flexible, pressure tight seal between the primary containment vessel and the diaphragm floor and is capable of accommodating differential thermal expansion between them.

The fatigue evaluation was performed using the cycles in FSAR Table 3A.4.1-3. The maximum cumulative usage factor is less than the fatigue limit per FSAR Table 3A.4.1-5. All events, including SRV actuations, for 60 years of operation are projected to remain below the containment cyclic basis from FSAR Table 3A.4.1-3. Consequently, the analysis of the diaphragm floor seal remains valid for the period of extended operation.

Disposition

The TLAA for fatigue of the containment diaphragm floor seal remains valid through the end of the period of extended operation.

A.2.2.6.6 ECCS Suction Strainers

The original Columbia ECCS suction strainers were replaced with a new strainer design constructed from cold-worked austenitic stainless steel. A linear elastic fracture mechanics analysis was performed to bound all the martensitic material in the suction strainer screens. A crack depth was assumed based on the depth of the Alpha Prime martensite in the strainer screen material.

Cyclic stresses were considered in the crack growth analysis of the suction strainers. The fatigue crack evaluation determined that the assumed cracks will not propagate to a critical size for the remaining life of the plant. The maximum computed stress intensity value (K) was less than that required to cause cracking in Alpha martensite formed in austenitic stainless steel.

The stress value conservatively included direct pressure and inertial components from SRV actuation, operating basis earthquake (OBE) loads, and SRV steam chugging. (See FSAR Table 3A.4.1-3.)

All events, including safety relief valve actuations, for 60 years of operation are projected to remain below the containment cyclic basis from FSAR Table 3A.4.1-3. Consequently, the analysis of the ECCS suction strainers remains valid for the period of extended operation.

Disposition

The TLAA for crack growth of the ECCS suction strainers remains valid through the end of the period of extended operation.

A.2.2.7 OTHER PLANT-SPECIFIC TIME-LIMITED AGING ANALYSES

The TLAAs that do not fit into any of the previous major categories are evaluated below.

A.2.2.7.1 Reactor Vessel Shell Indications

Two indications in the reactor vessel shell were identified using ultrasonic inspection methods during the 2005 inservice inspections. The indications were present in past inservice inspection examinations, but became rejectable under current ASME Section XI, IWB-3610 requirements. The rejected indications were evaluated and determined to be acceptable for continued service without repair, as reported to the NRC. The indications were evaluated per the guidelines of ASME Section XI, IWB-3610, which include acceptance criteria based on the

applied stress intensity factors, using conservative assumptions in the applied stresses to determine the stress intensity factors for comparison to Code allowables.

This conservative evaluation calculated a fatigue crack growth at the end of 33.1 EFPY vessel service life that is insignificant in comparison to the bounding initial crack size. It also determined that the applied stress intensity factor is well below the allowable stress intensity factor.

The calculation is based on time-limited assumptions of neutron fluence and SRV blowdown cycles for 40 years. While it is not expected that the applied stress intensity factor will exceed the allowable fracture toughness during the period of extended operation, cracking near the subject reactor vessel welds is managed by the Inservice Inspection (ISI) Program.

Energy Northwest will re-evaluate the indication based on the results of the 2015 inspection and either project this analysis through the period of extended operation or continue augmented inspections as required by the ASME code.

Disposition

Cracking of the reactor vessel shell near welds BG and BM will be adequately managed through the period of extended operation by the Inservice Inspection (ISI) Program.

A.2.2.7.2 Sacrificial Shield Wall

FSAR Section 3.8.3.6 provides a value of neutron fluence for the outside face of the sacrificial shield wall that is based on 40 years of plant operation. Projections done for 60 years of operation, including increase in fluence due to power uprate, determined that the estimated neutron fluence on the sacrificial shield wall will remain below the threshold for neutron damage of concrete and reinforcing steel. Therefore, the sacrificial shield wall can be expected to perform its radiation shielding function through the period of extended operation.

Disposition

The TLAA associated with the sacrificial shield wall fluence has been projected to the end of the period of extended operation.

A.2.2.7.3 Main Steam Flow Restrictor Erosion Analyses

The main steam line flow restrictors are designed to limit coolant flow rate from the reactor vessel (before the MSIVs are closed) to less than 200 percent of normal flow in the event of a main steam line break outside the 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. Since erosion is a time-related phenomenon, the analysis for the effect it has on the flow restrictors over the life of the plant is a TLAA. Cast stainless steel (SA351,

Type CF8) was selected for the steam flow restrictor material because it has excellent resistance to erosion-corrosion from high velocity steam.

The erosion of the main steam flow restrictors has been projected for the period of extended operation. The projection concludes that after 60 years of erosion on the main steam flow restrictors, the choked flow will still be less than 200 percent of normal flow. Therefore, the main steam flow restrictors will continue to perform their intended function and the existing accident radiological release analysis will remain valid for the period of extended operation.

Disposition

The TLAA for erosion of the main steam line flow restrictors has been projected to the end of the period of extended operation.

A.2.2.7.4 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.

Disposition

At least two years prior to the period of extended operation, Energy Northwest will install core plate wedges unless:

- a site-specific analysis is approved by the NRC that resolves core plate bolt loss of preload due to both stress relaxation and cracking, or
- an NRC approved method is developed to inspect the core plate bolts for cracking and a site-specific analysis for loss of preload due to stress relaxation of the core plate bolts is approved by the NRC.

A.2.2.7.5 Crane Load Cycle Limit

All in-scope cranes at Columbia were designed to Crane Manufacturers Association of America (CMAA) Specification 70, "*Specification for Electric Overhead Traveling Cranes*" which provides a design load cycle limit based on service class for the associated cranes. This load cycle limit for each crane was identified as a potential TLAA.

Disposition

To address this potential TLAA a 60-year projection of load cycles was developed for all cranes in the scope of license renewal and compared to the design load cycle limits of CMAA 70. For all cranes the 60-year projection of load cycles is within the applicable design load cycle limit of CMAA 70. Therefore, this TLAA remains valid for the period of extended operation.

A.3 REFERENCES

A.3-1 BWROG Report GE-NE-523-A71-0594-A, Rev 1, "Alternate BWR Feedwater Nozzle Inspection Requirements," May 2000

A.3-2 EPRI Report No. 1011838, "Recommendations for An Effective Flow Accelerated Corrosion Program (NSAC-202L-R3)," May 2006

A.3-3 EPRI TR-103834-P1-2, "Effects of Moisture on the Life of Power Plant Cables," August 1994