APPENDIX A

UPDATED FINAL SAFETY ANALYSIS REPORT SUPPLEMENT

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A.0 INTRODUCTION

This appendix provides the information to be submitted in the UFSAR Supplement as required by 10 CFR 54.21(d) [Reference 1.3-3] for the BVPS License Renewal Application (LRA). The LRA contains the technical information required by 10 CFR 54.21(a) and (c). Section 3 of the BVPS LRA identifies the programs and activities that will manage the effects of aging for the period of extended operation, and Appendix B provides descriptions of those programs and activities. Section 4 of the LRA documents the evaluations of time-limited aging analyses (TLAAs) for the period of extended operation. LRA Section 3, Section 4, and Appendix B have been used to prepare the program and activity descriptions for the BVPS UFSAR Supplement information in this Appendix.

This Appendix is divided into five sections:

- Section A.1 contains summary descriptions of the Unit 1 and Unit 2 programs used to manage the effects of aging during the period of extended operation;
- Section A.2 contains summary descriptions of the Unit 1 TLAAs during the period of extended operation;
- Section A.3 contains summary descriptions of the Unit 2 TLAAs during the period of extended operation;
- Section A.4 contains a listing of the Unit 1 commitments associated with license renewal; and,
- Section A.5 contains a listing of the Unit 2 commitments associated with license renewal.

The information presented in these five sections will be incorporated in the BVPS Unit 1 and Unit 2 UFSARs following issuance of the renewed operating licenses in accordance with 10 CFR 50.71(e) [Reference 1.3-1].

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A.1 SUMMARY DESCRIPTIONS OF AGING MANAGEMENT PROGRAMS

The license renewal integrated plant assessment and evaluation of time-limited aging analyses (TLAAs) identified existing and new aging management programs necessary to provide reasonable assurance that components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis (CLB) for the period of extended operation. This section describes the aging management programs and activities identified during the integrated plant assessment that will be required. Aging management programs will be implemented as identified in the list of license renewal commitments. The aging management programs associated with TLAAs are located in Sections A.1.14, A.1.27, and A.1.37.

Three elements of an effective aging management program that are common to all aging management programs are corrective actions, confirmation process, and administrative controls. These elements are included in the BVPS Quality Assurance (QA) Program, which implements the requirements of 10 CFR 50, Appendix B. Using the BVPS Corrective Action Program, adverse conditions are identified and categorized as conditions adverse to quality or significant conditions adverse to quality based on the significance and consequences of the specific problem identified. BVPS corrective actions, confirmation process, and administrative controls are consistent with NUREG-1801.

A.1.1 10 CFR PART 50, APPENDIX J PROGRAM

The BVPS 10 CFR Part 50, Appendix J Program monitors Containment leak rate. Containment leak rate tests are required to assure that (a) leakage through primary reactor Containment and systems and components penetrating primary Containment shall not exceed allowable values specified in technical specifications or associated bases and (b) periodic surveillance of reactor Containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of Containment, and systems and components penetrating primary Containment.

Appendix J provides two options, A and B, either of which can be chosen to meet the requirements of a Containment leak rate test program. BVPS uses option B, the performance-based approach. The Containment leak rate tests are performed in accordance with the guidelines contained in NRC Regulatory Guide 1.163, *Performance-Based Containment Leak-Testing Program* [Reference A.1-1] and NEI 94-01, *Industry Guidance for Implementing Performance-Based Options of 10 CFR Part 50 Appendix J* [Reference A.1-2].

A.1.2 ASME SECTION XI INSERVICE INSPECTION, SUBSECTIONS IWB, IWC, AND IWD PROGRAM

The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program is in accordance with ASME Boiler and Pressure Vessel Code, Section XI, Subsections IWB, IWC, and IWD, and is subject to the limitations and modifications of 10 CFR 50.55a. The program provides for condition monitoring of Class 1, 2, and 3 pressure-retaining components, including welds, pump casings, valve bodies, integral attachments, and pressure-retaining bolting. The program is updated as required by 10 CFR 50.55a.

The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program is augmented by the Water Chemistry Program (Section A.1.42) where applicable.

A.1.3 ASME SECTION XI, SUBSECTION IWE PROGRAM

The ASME Section XI, Subsection IWE Program is in accordance with ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWE. In conformance with 10 CFR 50.55a(g)(4)(ii), the BVPS ASME Section XI, Subsection IWE Program is updated during each successive 120-month inspection interval to comply with the requirements of the latest edition and addenda of the Code specified twelve months before the start of the inspection interval.

This program is implemented through plant procedures, which provide for inservice inspection of Class MC and metallic liners of Class CC components.

A.1.4 ASME SECTION XI, SUBSECTION IWF PROGRAM

The ASME Section XI, Subsection IWF Program is in accordance with ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWF. In conformance with 10 CFR 50.55a(g)(4)(ii), the BVPS ASME Section XI, Subsection IWF Program is updated during each successive 120-month inspection interval to comply with the requirements of the latest edition and addenda of the Code specified twelve months before the start of the inspection interval.

This program is implemented through plant procedures, which provide for visual examination of inservice inspection Class 1, 2, and 3 supports in accordance with the requirements of ASME Code Case N-491, *Alternate Rules for Examination of Class 1, 2, 3, and MC Component Supports of Light-Water Cooled Power Plants* [Reference A.1-5].

A.1.5 ASME Section XI, Subsection IWL Program

The ASME Section XI, Subsection IWL Program is in accordance with ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWL. In conformance with 10 CFR 50.55a(g)(4)(ii), the BVPS ASME Section XI, Subsection IWL Program is updated during each successive 120-month inspection interval to comply with the requirements of the latest edition and addenda of the Code specified twelve months before the start of the inspection interval.

The program consists of periodic visual inspections of the reinforced concrete Containment structures. An additional commitment requires that the inspectors be trained and certified in accordance with ASME, Section IX, Subsection IWL (1992 edition with the 1992 Addenda) standards. The BVPS concrete Containment structures do not utilize a post-tensioning system; therefore, the IWL requirements associated with a post-tensioning system are not applicable.

A.1.6 Bolting Integrity Program

The Bolting Integrity Program implements industry recommendations for a comprehensive bolting integrity program, as delineated in NUREG-1339, *Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants* [Reference A.1-6], and EPRI NP-5769, *Degradation and Failure of Bolting in Nuclear Power Plants* [Reference A.1-7]. Also, it implements industry recommendations for comprehensive bolting maintenance, as delineated in EPRI TR-104213, *Bolted Joint Maintenance & Application Guide* [Reference A.1-8], for pressure retaining bolting and structural bolting.

The program includes periodic inspection of closure bolting for indication of loss of preload, cracking, and loss of material due to corrosion, rust, etc. It also includes preventive measures to preclude or minimize loss of preload and cracking.

The program inspections are implemented through other Aging Management Programs listed as follows:

- ASME Section XI, *Inservice Inspection, Subsections IWB, IWC, & IWD Program*
- ASME Section XI, *Subsection IWE Program*
- ASME Section XI, *Subsection IWF Program*
- Structures Monitoring Program
- External Surfaces Monitoring Program

A.1.7 BORIC ACID CORROSION PROGRAM

The Boric Acid Corrosion Program manages loss of material due to borated water leakage by performing periodic visual inspections. The program relies in part on implementation of recommendations of NRC Generic Letter 88-05, *Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants* [Reference A.1-9].

The scope of the program inspections includes all systems that contain borated water, as well as components and systems that may be potentially impacted by borated water leakage. The program includes provisions for (a) determination of the principal location of leakage, (b) examination requirements and procedures for locating small leaks, and (c) engineering evaluations and corrective actions. If borated water leakage is discovered, either by program inspections or by other activities, it is evaluated and resolved using the Corrective Action Program.

A.1.8 BURIED PIPING AND TANKS INSPECTION PROGRAM

The Buried Piping and Tanks Inspection Program includes (a) preventive measures to mitigate corrosion, and, (b) inspections to manage the effects of corrosion on the pressure-retaining capability of buried steel and stainless steel components. Preventive measures are in accordance with standard industry practice for maintaining external coatings and wrappings. Buried components are inspected when excavated during maintenance or planned inspections. The program requires that, for each unit at BVPS, at least one opportunistic or focused inspection be performed and documented within the ten year period prior to, and within the ten year period after entering, the period of extended operation.

A.1.9 CLOSED-CYCLE COOLING WATER SYSTEM PROGRAM

The Closed-Cycle Cooling Water System Program includes: (1) preventive measures to minimize corrosion, and (2) periodic system and component performance testing and inspection to monitor the effects of corrosion and confirm that intended functions are met. This program manages loss of material, cracking, and reduction of heat transfer for components exposed to closed cooling water systems (Reactor / Primary Plant Component Cooling Water, Chilled Water, diesel-driven fire pump engine cooling water, Emergency Diesel Generator cooling water, Security Diesel Generator cooling water, Emergency Response Facility (ERF) diesel generator cooling water, and Unit 2 diesel-driven station standby air compressor engine cooling water).

These systems are closed cooling loops with controlled chemistry, consistent with the NUREG-1801 [Reference 1.3-5] description of a closed cycle cooling water system. The adequacy of chemistry control is confirmed on a routine basis by sampling and ensuring contaminants and additives are within established limits, and by equipment performance monitoring to identify aging effects. These chemistry activities are controlled using BVPS procedures and processes and are based on EPRI guidance for closed cooling water chemistry located in EPRI 1007820 (EPRI 107396, Rev. 1) [Reference A.1-10].

A.1.10 ELECTRICAL CABLE CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS ONE-TIME INSPECTION PROGRAM

The Electrical Cable Connections not Subject to 10 CFR 50.49 Environmental Qualification Requirements One-Time Inspection Program is a one-time inspection program that inspects and tests the metallic parts of the cable connection. A representative sample of electrical cable connection population subject to aging management review is inspected or tested. Electrical connections covered under the Environmental Qualification (EQ) Program (Section A.1.14), or connections inspected or tested as part of a preventive maintenance program, are excluded from aging management review.

This sampling program provides a one-time inspection to verify that the loosening of bolted connections due to thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, and oxidation is not an aging issue that requires a periodic aging management program. The design of these connections accounts for the stresses associated with ohmic heating, thermal cycling, and dissimilar metal connections. Therefore, these stressors or mechanisms should not be a significant aging issue. However, confirmation of the lack of aging effects is required. The factors considered for sample selection are application (medium and low voltage), circuit loading (high loading), and location (high temperature, high humidity, vibration, etc.). The technical basis for the sample selection will be documented. Any unacceptable conditions found during the inspection will be evaluated through the Corrective Action Program.

For Unit 2 only, the metallic parts of metal enclosed bus connections are managed by the Metal Enclosed Bus Program (Unit 2 only) (Section A.1.26), and are therefore not included within the scope of the program. There is no in-scope metal enclosed bus at Unit 1.

A.1.11 ELECTRICAL CABLES AND CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS PROGRAM

The Electrical Cables And Connections Not Subject To 10 CFR 50.49 Environmental Qualification Requirements Program provides reasonable assurance that intended functions of insulated cables and connections exposed to adverse localized environments caused by heat, radiation and moisture can be maintained consistent with the current licensing basis through the period of extended operation. An "adverse localized environment" is an environment that is significantly more severe than the specified service condition for the insulated cable or connection.

A representative sample of accessible insulated cables and connections within the scope of license renewal and located in adverse localized environments will be visually inspected at least once every 10 years for cable and connection jacket surface anomalies such as embrittlement, discoloration, cracking or surface contamination. The program requires the first inspection to be completed prior to entering the period of extended operation. The technical basis for sampling is derived from the guidance provided by applicable industry documents.

A.1.12 ELECTRICAL CABLES AND CONNECTIONS NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS USED IN INSTRUMENTATION CIRCUITS PROGRAM

The Electrical Cables And Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program demonstrates that sensitive (high voltage – low current applications) instrument cables and connections susceptible to aging effects caused by exposure to adverse localized environments caused by heat, radiation, and moisture are adequately managed so that there is reasonable assurance that the cables and connections will perform their intended function in accordance with the current licensing basis during the period of extended operation. An “adverse localized environment” is an environment that is significantly more severe than the specified service condition for the cable. This aging management program requires a review of non-EQ instrumentation circuit calibration results at least once every ten years, with the initial performance of this program to occur prior to the period of extended operation. BVPS will incorporate into the program the appropriate technical information and guidance provided in industry documents.

A.1.13 ELECTRICAL WOODEN POLES/STRUCTURES INSPECTION PROGRAM (UNIT 2 ONLY)

The Electrical Wooden Poles/Structures Inspection Program manages aging effects for wooden poles subject to aging management, such as insect and woodpecker damage, reduced circumference, and moisture intrusion. Appropriate aging management methods include pole sounding, pole boring, and underground inspection.

A.1.14 ENVIRONMENTAL QUALIFICATION (EQ) OF ELECTRICAL COMPONENTS PROGRAM

The Environmental Qualification (EQ) of Electrical Components Program manages the effects of thermal, radiation, and cyclic aging through the use of aging evaluations based on 10 CFR 50.49 qualification methods. As required by 10 CFR 50.49, environmental qualification program components not qualified for the current license term are refurbished, replaced, or their qualification extended prior to reaching the aging limits established in the evaluations. Aging evaluations for environmental qualification program components are time-limited aging analyses (TLAAs) for license renewal.

A.1.15 EXTERNAL SURFACES MONITORING PROGRAM

The External Surfaces Monitoring Program is based on system inspections and walkdowns. This program consists of periodic inspections to monitor the external surfaces of in-scope steel components and other metal components for material degradation and leakage, and periodic inspection of in-scope elastomer components for hardening, loss of strength or cracking through physical manipulation. The program will also require inspection of the Emergency Response Facility (ERF) diesel generator jacket water radiator fins for build-up of dust, dirt and debris. Additionally, the program is credited with managing aging effects of internal surfaces, for situations in which material and environment combinations are the same for internal and external surfaces such that external surface condition is representative of internal surface condition.

Loss of material due to boric acid corrosion is managed by the Boric Acid Corrosion Program.

A.1.16 FIRE PROTECTION PROGRAM

The Fire Protection Program is a condition monitoring and performance monitoring program, comprised of tests and inspections that follow the applicable National Fire Protection Association (NFPA) recommendations. The Fire Protection Program manages the aging effects on fire barrier penetration seals; fire barrier walls, ceilings and floors; fire wraps and fire rated doors (automatic and manual) that perform a current licensing basis fire barrier intended function. It also manages the aging effects on the diesel engine-driven fire pump fuel oil supply line. The Fire Protection Program also manages the aging effects on the halon and carbon dioxide fire suppression systems.

A.1.17 FIRE WATER SYSTEM PROGRAM

The Fire Water System Program applies to the water filled fire protection subsystems consisting of sprinklers, nozzles, fittings, valves, hydrants, hose stations, standpipes, tanks, and aboveground and underground piping and components that are tested in accordance with applicable National Fire Protection Association (NFPA) codes and standards. This program is credited with managing loss of material and reduction of heat transfer (reduction of heat transfer applies to the diesel-driven fire pump jacket water and oil coolers) for the water-filled Fire Protection Systems. Program activities include periodic inspection and hydro-testing of hydrants and hose stations, performing sprinkler head inspections, and conducting system flow tests. These tests and inspections follow applicable NFPA guidelines as well as recommendations from the fire insurance carrier. Such testing assures functionality of the systems. Also, many of these systems are normally maintained at required operating pressure and monitored such that leakage resulting in loss of system pressure is immediately detected and corrective actions initiated.

All sprinkler heads will be replaced, or a sample population will be inspected using the guidance of NFPA 25, *Standard for the Inspection, Testing and Maintenance of Water-Based Fire Protection Systems* [Reference A.1-11]. NFPA 25, Section 5.3.1.1.1 states that "where sprinklers have been in place for 50 years, they shall be replaced or representative samples from one or more sample areas shall be submitted to a recognized testing laboratory for field service testing." If the sampling method is chosen, NFPA 25 also contains guidance to perform this sampling every 10 years after initial field service testing.

A.1.18 FLOW-ACCELERATED CORROSION PROGRAM

The Flow-Accelerated Corrosion Program is based on EPRI guidelines in NSAC-202L-R2, *Recommendations for an Effective Flow Accelerated Corrosion Program* [Reference A.1-12]. The program predicts, detects, and monitors wall thinning in piping, valve bodies, and other in-line components. Analytical evaluations and periodic examinations of locations that are most susceptible to wall thinning due to flow-accelerated corrosion are used to predict the amount of wall thinning. The program includes analyses to determine critical locations. Initial inspections are performed to determine the extent of thinning at these critical locations, and follow-up inspections are used to confirm the predictions. Inspections are performed using ultrasonic, visual or other approved inspection techniques capable of detecting wall thinning. Repairs and replacements are performed as necessary.

A.1.19 FLUX THIMBLE TUBE INSPECTION PROGRAM

The Flux Thimble Tube Inspection Program serves to identify loss of material due to wear prior to leakage by monitoring for and predicting unacceptable levels of wall thinning in the Movable

Incore Detector System Flux Thimble Tubes, which serve as a Reactor Coolant System pressure boundary. The program implements the recommendations of NRC IE Bulletin 88-09, *Thimble Tube Thinning in Westinghouse Reactors* [Reference A.1-13].

The main attribute of the program is periodic nondestructive examination of the flux thimble tubes which provides actual values of existing tube wall thinning. This information provides the basis for an extrapolation to determine when tube wall thinning will progress to an unacceptable value. Based on this prediction, preemptive actions are taken to reposition, replace or isolate the affected thimble tube prior to a pressure boundary failure.

A.1.20 FUEL OIL CHEMISTRY PROGRAM

The Fuel Oil Chemistry Program is a mitigation and condition monitoring program which manages aging effects of the internal surfaces of oil storage tanks and associated components in systems that contain diesel fuel oil. The program includes (a) surveillance and monitoring procedures for maintaining diesel fuel oil quality by controlling contaminants in accordance with ASTM Standards D 975, D 1796, D 2276 and D 4057; (b) periodic sampling of fuel oil tanks and new fuel oil shipments for the presence of water and contaminants, and draining of any accumulated water from the tanks; (c) sampling of fuel oil tanks and new fuel oil shipments for numerous other factors such as sediment, viscosity, and flash point; (d) periodic or conditional visual inspection of internal surfaces or wall thickness measurements (e.g., ultrasonic testing) of tanks.

The One-Time Inspection Program (Section A.1.30) will be used to verify the effectiveness of the Fuel Oil Chemistry Program.

A.1.21 INACCESSIBLE MEDIUM-VOLTAGE CABLES NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS PROGRAM

The Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program demonstrates that inaccessible, non-EQ medium-voltage cables susceptible to aging effects caused by moisture and voltage stress are adequately managed so that there is reasonable assurance that the cables will perform their intended function in accordance with the current licensing basis during the period of extended operation.

In this aging management program, periodic actions are taken, at least once every two years, to prevent cables from being exposed to significant moisture, such as inspecting for water collection in cable manholes and conduit, and draining water, as needed. In-scope, medium-voltage cables exposed to significant moisture and significant voltage are tested to provide an indication of the condition of the conductor insulation. The specific type of test performed is determined prior to the initial test, and is a proven test for detecting deterioration of the insulation system due to wetting, such as power factor, partial discharge, or other testing that is state-of-the-art at the time the test is performed. Testing is conducted at least once every 10 years, with initial testing completed prior to the period of extended operation.

A.1.22 Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program

The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program consists of inspections of the internal surfaces of piping, piping components, ducting and other components within the scope of license renewal that are not covered by other aging management programs. The internal inspections are performed during periodic system and component surveillances or during the performance of maintenance activities when the surfaces are made accessible for visual inspection. These inspections will assure that existing environmental conditions are not causing material degradation that could result in a loss of intended function.

A.1.23 Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems Program

The Inspection of Overhead Heavy Load & Light Load (Related To Refueling) Handling Systems Program manages loss of material of structural components for heavy load and fuel handling components within the scope of license renewal and subject to aging management. The program is implemented through plant procedures and preventive maintenance activities that provide for visual inspections of the in-scope load handling components.

The inspections are focused on structural components that make up the bridge, trolley, and rails of the cranes and hoists. These cranes and hoists also comply with the maintenance rule requirements provided in 10 CFR 50.65.

Overhead heavy load cranes are controlled in accordance with the guidance provided in NUREG-0612, *Control of Heavy Loads at Nuclear Power Plants* [Reference A.1-14].

A.1.24 Lubricating Oil Analysis Program

The purpose of the Lubricating Oil Analysis Program is to ensure the lubricating oil environment for in-scope mechanical systems is maintained to the required quality. The program monitors and controls abnormal levels of contaminants (primarily water and particulates) for in-scope components in the lubricating oil systems, thereby preserving an environment that is not conducive to loss of material, cracking, or reduction of heat transfer.

The One-Time Inspection Program (Section A.1.30) will be used to verify the effectiveness of the Lubricating Oil Analysis Program.

A.1.25 MASONRY WALL PROGRAM

The Masonry Wall Program manages the aging effects of masonry walls that are within the scope of License Renewal and subject to aging management review. The program consists of visual inspections to identify cracks in masonry walls and ensure the sound condition of structural steel supports and bracing associated with masonry walls.

Masonry walls in close proximity to, or having attachments from, safety-related systems or components are inspected in response to NRC IE Bulletin 80-11, *Masonry Wall Design* [Reference A.1-15], and NRC Information Notice 87-67, *Lessons Learned from Regional Inspections of Licensee Actions in Response to IE Bulletin 80-11* [Reference A.1-16]. These inspections consist of a visual examination by qualified personnel to ensure that the evaluation basis for these walls remains valid through the period of extended operation.

In addition, a general visual inspection is performed on both safety-related and nonsafety-related masonry walls that are within the scope of license renewal. These inspections are implemented by the Structures Monitoring Program (Section A.1.39) and consist of visual inspection for cracking in joints, deterioration of penetrations, missing or broken blocks, missing mortar, and general mechanical soundness of steel supports.

A.1.26 METAL ENCLOSED BUS PROGRAM (UNIT 2 ONLY)

The Metal Enclosed Bus Program is applicable only to the Unit 2 480-VAC Metal Enclosed Bus Feeders to the Emergency Substations (2-8 and 2-9). The program requires visual inspections of in-scope metal enclosed bus internal surfaces for aging degradation of insulating and conductive components. The visual inspection also identifies evidence of foreign debris, excessive dust buildup, or moisture intrusion. The bus insulating system, including the internal supports, is visually inspected for structural integrity and signs of aging degradation. A sample of accessible bolted connections are checked for loose connection using thermography. Inspections are completed prior to the period of extended operation and every 10 years thereafter.

A.1.27 METAL FATIGUE OF REACTOR COOLANT PRESSURE BOUNDARY PROGRAM

The Metal Fatigue of Reactor Coolant Pressure Boundary Program is a time-limited aging analysis (TLAA) program that uses preventive measures to mitigate fatigue cracking caused by anticipated cyclic strains in metal components of the reactor coolant pressure boundary. The preventive measures consist of monitoring and tracking critical thermal and pressure transients for Reactor Coolant System components to prevent the fatigue design limit from being exceeded. Critical transients are the subset of the design transients that are expected to approach or exceed the number of design cycles during the sixty year operating life of the units. Prior to exceeding the fatigue design limit, preventive and/or corrective actions are triggered by the program.

In addition, environmental effects are evaluated in accordance with NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves for Selected Nuclear Power Plant Components* [Reference A.1-17] and the guidance of EPRI Technical Report MRP-47, *Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application* [Reference A.1-18]. Selected components are evaluated using material specific guidance presented in NUREG/CR-6583, *Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low Alloy Steels* [Reference A.1-19], and in NUREG/CR-5704, *Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels* [Reference A.1-20].

A.1.28 NICKEL-ALLOY NOZZLES AND PENETRATIONS PROGRAM

For the Nickel-Alloy Nozzles and Penetrations Program, regarding activities for managing the aging of nickel-alloy and nickel-alloy clad components susceptible to primary water stress corrosion cracking - PWSCC (other than upper reactor vessel closure head nozzles and penetrations), BVPS commits to develop a plant-specific aging management program that will implement applicable:

1. NRC Orders, Bulletins and Generic Letters, and,
2. staff-accepted industry guidelines.

A.1.29 NICKEL-ALLOY PENETRATION NOZZLES WELDED TO THE UPPER REACTOR VESSEL CLOSURE HEADS PROGRAM

The Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Head Program manages cracking due to primary water stress corrosion cracking in nickel-alloy vessel head penetration nozzles. The program scope includes the reactor vessel closure head, upper vessel head penetration nozzles, and associated welds. The program also is used in conjunction with the Boric Acid Corrosion Program to examine the reactor vessel upper head for any loss of material due to boric acid wastage. This program was developed in response to NRC Order EA-03-009, *Issuance of Order Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors* [Reference A.1-21], and First Revised Order EA-03-009, *Issuance of First Revised NRC Order (EA-03-009) Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors*

[Reference A.1-22]. Detection of cracking is accomplished through implementation of a combination of bare metal visual examination (external surface of head) and non-visual examination techniques.

A.1.30 ONE-TIME INSPECTION PROGRAM

The One-Time Inspection Program requires one-time inspections to verify effectiveness of the Water Chemistry Program (Section A.1.42), the Fuel Oil Chemistry Program (Section A.1.20), and the Lubricating Oil Analysis Program (Section A.1.24). One-time inspections may be needed to address concerns for potentially long incubation periods for certain aging effects on structures and components. There are cases where either (a) an aging effect is not expected to occur but there is insufficient data to completely rule it out, or (b) an aging effect is expected to progress very slowly. For these cases, there will be confirmation that either the aging effect is indeed not occurring, or the aging effect is occurring very slowly as not to affect the component or structure intended function during the extended period of operation. The one-time inspections provide additional assurance that, either aging is not occurring, or aging is so insignificant that an aging management program is not warranted.

The elements of the program include:

- Determination of a representative sample size based on an assessment of materials of fabrication, environment, plausible aging effects, and operating experience;
- Identification of the inspection locations in the system or component based on the aging effect, or areas susceptible to concentration of agents that promote certain aging effects;
- Determination of the examination technique, including acceptance criteria that would be effective in managing the aging effect for which the component is examined; and,
- Evaluation of the need for follow-up examinations to monitor the progression of any aging degradation.

In addition to verifying program effectiveness, the program is used to verify loss of material is not occurring in the following components:

- Steam generator feedwater ring; and,
- Selected bottoms of tanks that sit on concrete pads (by volumetric examination).

When evidence of an aging effect is revealed by a one-time inspection, the routine evaluation of the inspection results would identify appropriate corrective actions.

A.1.31 ONE-TIME INSPECTION OF ASME CODE CLASS 1 SMALL BORE PIPING PROGRAM

The One-Time Inspection of ASME Code Class 1 Small-Bore Piping Program manages cracking of stainless steel ASME Code Class 1 piping less than 4 inches nominal pipe size (NPS 4), which includes pipes, fittings, and branch connections. The program will manage this aging effect by performing volumetric examinations for selected ASME Code Class 1 small-bore butt welds.

Should evidence of significant aging be revealed by the one-time inspection, periodic inspection will be proposed, as managed by a plant-specific aging management program.

A.1.32 OPEN-CYCLE COOLING WATER SYSTEM PROGRAM

The Open-Cycle Cooling Water System Program implements the site commitments to NRC Generic Letter 89-13, *Service Water System Problems Affecting Safety-Related Equipment* [Reference A.1-23], including Supplement 1. This program manages the aging effects on the open-cycle cooling water systems such that the systems will be able to fulfill their intended function during the period of extended operation. The program includes surveillance and control techniques to manage aging effects caused by biofouling, corrosion, erosion, protective coating failures, and silting in the River Water (Unit 1 only) / Service Water (Unit 2 only) Systems or structures and components serviced by the systems.

A.1.33 PWR VESSEL INTERNALS PROGRAM

For the PWR Vessel Internals Program, regarding activities for managing the aging of Reactor Vessel internal components and structures, BVPS provided in Table A.4-1 (Unit 1 only) and Table A.5-1 (Unit 2 only) commitments to:

1. Participate in the industry programs applicable to BVPS for investigating and managing aging effects on reactor internals;
2. Evaluate and implement the results of the industry programs as applicable to the BVPS reactor internals; and,
3. Upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for the BVPS reactor internals to the NRC for review and approval.

A.1.34 Reactor Head Closure Studs Program

The Reactor Head Closure Studs Program at BVPS Unit 1 and Unit 2 manages the aging effects of the reactor head closure studs, nuts, washers and associated Reactor Vessel flange threads. The program is part of the BVPS ASME Code Section XI Inservice Inspection Program. The examinations are performed in accordance with Code Section XI, 1989 edition with no Addenda. The Program is updated periodically as required by 10 CFR 50.55a. The program preventive measures are consistent with the recommendations of Regulatory Guide 1.65, *Materials and Inspections for Reactor Vessel Closure Studs* [Reference A.1-24].

A.1.35 Reactor Vessel Integrity Program

The Reactor Vessel Integrity Program manages loss of fracture toughness due to neutron embrittlement in reactor materials exposed to neutron fluence exceeding 1.0E+17 n/cm^2 (E>1.0 MeV). The program is based on 10 CFR 50, Appendix H, *Reactor Vessel Material Surveillance Requirements*, and ASTM Standard E 185-82, *Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels* [Reference A.1-25] (incorporated by reference into 10 CFR 50, Appendix H). Capsules are periodically removed during the course of plant operating life. Neutron embrittlement is evaluated through surveillance capsule testing and evaluation, fluence calculations and monitoring of effective full power years (EFPYs). Best-estimate values of Reactor Vessel accumulated neutron fluence are determined utilizing analytical models that satisfy the guidance contained in NRC Regulatory Guide 1.190, *Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence* [Reference A.1-26]. Data resulting from the program is used to:

- Determine pressure-temperature limits, minimum temperature requirements, and end-of-life Charpy upper-shelf energy (C_VUSE) in accordance with the requirements of 10 CFR 50 Appendix G, *Fracture Toughness Requirements*; and,
- Determine end-of-life RT_{PTS} values in accordance with 10 CFR 50.61, *Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock*.

The Reactor Vessel Integrity Program provides guidance for removal and testing or storage of material specimen capsules. All capsules that have been withdrawn were tested and stored. Standby capsules at Unit 1 and Unit 2 will be available for future testing. Standby capsules from each unit will be removed from the vessel when the neutron fluences are approximately equivalent to the expected vessel wall neutron fluence at 60 years of operation (corrected for lead and capacity factors).

In addition, the Reactor Vessel Integrity Program implements flux reduction programs as required by 10 CFR 50.61.

A.1.36 SELECTIVE LEACHING OF MATERIALS PROGRAM

The Selective Leaching of Materials Inspection Program includes a one-time visual inspection and hardness examination of selected components that are susceptible to selective leaching. The program scope includes components and commodities (such as piping, pump casings, valve bodies and heat exchanger components) made of copper alloys with zinc content greater than 15% or gray cast iron which are exposed to a raw water, treated water, air, condensation, or soil environment.

This program determines whether selective leaching is occurring for selected components. Should evidence of significant aging be revealed by the one-time inspection or previous operating experience, the Corrective Action Program is used for the unacceptable inspection findings. The resolution will include evaluation for expansion of the inspection sample size, locations, and frequency.

A.1.37 SETTLEMENT MONITORING PROGRAM (UNIT 2 ONLY)

The Settlement Monitoring Program (Unit 2 only) is an existing plant-specific condition monitoring program for structures and piping that are within the scope of license renewal. The program monitors the settlement of structures to prevent stresses in the structures or piping from increasing beyond analyzed stress levels. The analyses of the structures and piping addressed by the program are time-limited aging analyses (TLAAs) discussed in Section A.3.6.3. The Settlement Monitoring Program ensures that the current 40-year settlement assumptions in the Unit 2 pipe stress analyses are maintained for the period of extended operation.

A.1.38 STEAM GENERATOR TUBE INTEGRITY PROGRAM

The Steam Generator Tube Integrity Program is based on NEI 97-06, *Steam Generator Program Guidelines* [Reference A.1-27]. The Steam Generator Tube Integrity Program is credited for aging management of the tubes, tube plugs, tube supports, and the secondary-side internal components whose failure could prevent the steam generator from fulfilling its intended safety function. The program includes performance criteria that are intended to provide assurance that steam generator tube integrity is being maintained consistent with the plant's licensing basis, and provides guidance for monitoring and maintaining the tubes to provide assurance that the performance criteria are met at all times between scheduled inspections of the tubes.

The Steam Generator Tube Integrity Program provides the requirements for inspection activities for the detection of flaws in tubes, plugs, tube supports, and secondary-side internal components needed to maintain tube integrity. Degradation assessments identify both potential and existing degradation mechanisms. Inservice inspections (i.e., eddy current testing, ultrasonic testing and visual inspections) are used for the detection of flaws. Condition monitoring compares the inspection results against performance criteria, and an operational assessment provides a prediction of tube conditions to ensure that the performance criteria will not be exceeded during the next operating cycle. Primary to secondary leakage is continually monitored during operation.

A.1.39 STRUCTURES MONITORING PROGRAM

The Structures Monitoring Program implements the requirements of 10 CFR 50.65, *Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants* (the Maintenance Rule), using the guidance of NUMARC 93-01, *Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants* [Reference A.1-28] and Regulatory Guide 1.160, *Monitoring the Effectiveness of Maintenance at Nuclear Power Plants* [Reference A.1-29].

The program relies on periodic visual inspections to monitor the condition of structures and structural components so that intended functions are maintained through the period of extended operation.

A.1.40 THERMAL AGING AND NEUTRON IRRADIATION EMBRITTLEMENT OF CAST AUSTENITIC STAINLESS STEEL (CASS) PROGRAM

The Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program inpects Reactor Vessel Internals in accordance with ASME Code Section XI, Subsection IWB, Category B-N-3. This inspection is augmented to detect the effects of loss of fracture toughness due to thermal aging and neutron irradiation embrittlement of CASS components. The program includes identification of the limiting susceptible components from the standpoint of thermal aging susceptibility, neutron fluence, and cracking. For each identified component, aging management is accomplished through either a supplemental examination or a component-specific evaluation, including a mechanical loading assessment.

BVPS participates in the EPRI Materials Reliability Project established to investigate the impacts of aging on PWR vessel internal components. The results of this project provide additional bases for the inspections and evaluations performed under this program.

A.1.41 THERMAL AGING EMBRITTLEMENT OF CAST AUSTENITIC STAINLESS STEEL (CASS) PROGRAM

The Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program inspects Reactor Coolant System components in accordance with the ASME Boiler and Pressure Vessel

Code, Section XI. The ASME Section XI inspection is augmented to detect the effects of loss of fracture toughness due to thermal aging embrittlement of cast austenitic stainless steel components. This program includes a determination of the susceptibility of the subject cast austenitic stainless steel components to thermal aging embrittlement based on casting method, molybdenum content, and percent ferrite. For potentially susceptible components, aging management is accomplished utilizing additional inspections or a component-specific flaw tolerance evaluation. Additional inspections or evaluations are not required for components that are determined not to be susceptible to thermal aging embrittlement. Screening for susceptibility to thermal aging embrittlement is not required for pump casings and valve bodies. The existing ASME Section XI inspection requirements, including the alternative requirements of ASME Code Case N-481 *Alternate Examination Requirements for Cast Austenitic Pump Casings*, [Reference A.1-30], are adequate for all pump casings and valve bodies.

In addition, cast austenitic stainless steel components that are not part of the reactor coolant pressure boundary, but that have service conditions above 250º C (> 482º F), are included in this program. These components will be inspected, evaluated, or replaced as appropriate if screening determines they are susceptible to thermal aging embrittlement. The screening exclusion (pump casings and valve bodies) is not applicable to these components.

A.1.42 WATER CHEMISTRY PROGRAM

The main objective of the Primary and Secondary Water Chemistry Program is to mitigate damage caused by corrosion and stress corrosion cracking. The Water Chemistry Program relies on monitoring and control of water chemistry based on EPRI TR-105714, Rev. 5 (TR-1002884), *PWR Primary Water Chemistry Guidelines* [Reference A.1-31], and EPRI TR-102134, Rev. 6 (TR-1008224), *PWR Secondary Water Chemistry Guidelines* [Reference A.1-32].

The One-Time Inspection Program (XI.M32) will be used to verify the effectiveness of the Water Chemistry Program for the circumstances identified in NUREG-1801 that require augmentation of the Water Chemistry Program.

A.1.43 APPENDIX A.1 REFERENCES

A.1-1 Regulatory Guide 1.163, *Performance-Based Containment Leak-Testing Program,* September 1995.

A.1-2 NEI 94-01, *Industry Guidance for Implementing Performance-Based Options of 10 CFR Part 50 Appendix J,* Rev. 0.

A.1-3 WCAP 14572, Rev. 1-NP-A, Addenda 1, *Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report.*

A.1-4 NRC letter dated 4/9/04, *Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and 2) - Risk-Informed Inservice Inspection (RI-ISI) Program.*

A.1-5 ASME Code Case N-491, *Alternate Rules for Examination of Class 1, 2, 3, and MC Component Supports of Light-Water Cooled Power Plants,* March 28, 2000.

A.1-6 NUREG-1339, *Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants,* October 17, 1991.

A.1-7 EPRI NP-5769, *Degradation and Failure of Bolting in Nuclear Power Plants,* May 5, 1988.

A.1-8 EPRI TR-104213, *Bolted Joint Maintenance & Application Guide,* December 1, 1995.

A.1-9 NRC Generic Letter 88-05, *Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants,* March 17, 1988.

A.1-10 EPRI 1007820 (EPRI TR-107396, Rev. 1), *Closed Cooling Water Chemistry Guideline*, Rev. 1.

A.1-11 National Fire Protection Association NFPA 25, *Standard for the Inspection, Testing and Maintenance of Water-Based Fire Protection Systems,* 2002 Edition.

A.1-12 NSAC-202L-R2, *Recommendations for an Effective Flow Accelerated Corrosion Program,* April 1999.

A.1-13 NRC IE Bulletin 88-09, *Thimble Tube Thinning in Westinghouse Reactors,* July 26, 1988.

A.1-14 NUREG-0612, *Control of Heavy Loads at Nuclear Power Plants,* July 1980.

A.1-15 NRC IE Bulletin 80-11, *Masonry Wall Design,* May 8, 1980.

A.1-16 NRC Information Notice 87-67, *Lessons Learned from Regional Inspections of Licensee Actions in Response to IE Bulletin 80-11,* December 31, 1987.

A.1-17 NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves for Selected Nuclear Power Plant Components,* February 28, 1995.

A.1-18 EPRI Technical Report MRP-47, *Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application,* September 1, 2005.

A.1-19 NUREG/CR-6583, *Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low Alloy Steels,* February 1998.

A.1-20 NUREG/CR-5704, *Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels,* April 1999.

A.1-21 NRC Order EA 03-009, *Issuance of Order Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors,* February 11, 2003.

A.1-22 NRC First Revised Order EA-03-009, *Issuance of Revised Order EA-09-003 Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors,* February 11, 2004.

A.1-23 NRC Generic Letter 89-13, *Service Water System Problems Affecting Safety-Related Equipment,* including Supplement 1, July 18, 1989.

A.1-24 Regulatory Guide 1.65, *Materials and Inspections for Reactor Vessel Closure Studs,* October 1973.

A.1-25 ASTM Standard E 185-82, *Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels,* June 2002.

A.1-26 Regulatory Guide 1.190, *Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence*, March 2001.

A.1-27 NEI 97-06, *Steam Generator Program Guidelines,* Rev. 2, May 2005.

A.1-28 NUMARC 93-01, *Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,* Rev. 3, October 8, 1999.

A.1-29 Regulatory Guide 1.160, *Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,* Rev. 2, March 1997.

A.1-30 ASME Code Case N-481, *Alternate Examination Requirements for Cast Austenitic Pump Casings,* May 20, 1998.

A.1-31 EPRI TR-105714, Rev. 5 (TR-1002884), *PWR Primary Water Chemistry Guidelines*

A.1-32 EPRI TR-102134, Rev. 6 (TR-1008224), *PWR Secondary Water Chemistry Guidelines*

A.2 EVALUATION SUMMARIES OF UNIT 1 TIME-LIMITED AGING ANALYSES

A.2.1 INTRODUCTION

Time-limited aging analyses (TLAAs) are defined in 10 CFR 54.3 [Reference A.2-3] as:

Time-limited aging analyses, for the purposes of this part, are those licensee calculations and analyses that:

1. *Involve systems, structures, and components within the scope of license renewal, as delineated in §54.4(a);*
2. *Consider the effects of aging;*
3. *Involve time-limited assumptions defined by the current operating term, for example, 40 years;*
4. *Were determined to be relevant by the licensee in making a safety determination;*
5. *Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, and component to perform its intended functions, as delineated in §54.4(b); and*
6. *Are contained or incorporated by reference in the CLB.*

Once identified, TLAAs must be evaluated and dispositioned as described in the following section of 10 CFR 54:

§54.21 *Contents of application -- technical information.*

(c) An evaluation of time-limited aging analyses.

1. *A list of time-limited aging analyses, as defined in §54.3, must be provided. The applicant shall demonstrate that —*
   - *(i). The analyses remain valid for the period of extended operation;*
   - *(ii). The analyses have been projected to the end of the period of extended operation; or*
   - *(iii). The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.*

This chapter provides a summary of the TLAAs identified in the BVPS License Renewal Application, and includes the following topics:

- Reactor Vessel Neutron Embrittlement (Section A.2.2)

- Metal Fatigue (Section A.2.3)
- Environmental Qualification (EQ) of Electric Equipment (Section A.2.4)
- Containment Liner Plate, Metal Containment, and Penetrations Fatigue (Section A.2.5)
- Other Plant-Specific Time-Limited Aging Analyses (Section A.2.6)
- Appendix A.2 References (Section A.2.7)

A.2.2 Reactor Vessel Neutron Embrittlement

Analyses that address the effects of neutron irradiation embrittlement of the Reactor Vessels and were identified as TLAAs are summarized in the following sections:

- Neutron Fluence Values (Section A.2.2.1)
- Pressurized Thermal Shock (Section A.2.2.2)
- Charpy Upper Shelf Energy (Section A.2.2.3)
- Pressure-Temperature Limits (Section A.2.2.4)

A.2.2.1 Neutron Fluence Values

Loss of fracture toughness is an aging effect caused by the neutron embrittlement aging mechanism that results from prolonged exposure to neutron radiation. This process results in increased tensile strength and hardness of the material with reduced toughness. The rate of neutron exposure is defined as neutron flux, and the cumulative degree of exposure over time is defined as neutron fluence. As neutron embrittlement progresses, the toughness/temperature curve shifts down (lower fracture toughness), and the curve shifts to the right (brittle/ductile transition temperature increases).

In the spring of 2000, Surveillance Capsule Y was pulled and the analysis was documented in WCAP-15571, *Analysis of Capsule Y from First Energy Company Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program* [Reference A.2-4]. For license renewal, WCAP-15571 Supplement 1, *Analysis of Capsule Y from First Energy Company Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program* [Reference A.2-5], documents the end-of-license-extended (EOLE) analysis for neutron fluence values.

The fluence values were projected using ENDF/B-VI cross sections, are based on the results of the Capsule Y analysis, and comply with Regulatory Guide 1.190, *Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence* [Reference A.2-6].

The fluence projections include fuel cycle-specific calculated neutron exposures at the end of Cycle 17 (February 2006), as well as future projections to the end of Cycle 18 (September 2007) and for several intervals extending to 54 effective full power years (EFPY). The calculations account for a core power uprate from 2689 megawatts-thermal (MWt) to 2900 MWt at the onset of Cycle 18. Neutron exposure projections beyond the end of Cycle 17 were based on the spatial power distributions and associated plant characteristics of Cycle 18 in conjunction with the uprated power level.

A.2.2.2 Pressurized Thermal Shock

In the spring of 2000, Surveillance Capsule Y was pulled and the analysis was documented in WCAP-15571 [Reference A.2-4]. For license renewal, WCAP-15571 Supplement 1 [Reference A.2-5] documents the EOLE analysis for pressurized thermal shock (PTS).

Using the prescribed PTS Rule (10 CFR 50.61 [Reference A.2-7]) methodology, reference temperature for pressurized thermal shock (RT_{PTS}) values were generated for beltline and extended beltline region materials of the BVPS Unit 1 Reactor Vessel for fluence values at EOLE (54 EFPY). The projected RT_{PTS} values for EOLE (54 EFPY) meet the 10 CFR 50.61 screening criteria for beltline and extended beltline materials, with the exception of lower shell plate B6903-1 (heat C6317-1), which slightly exceeds the criteria with a RT_{PTS} of 275.7°F. The screening limit of 270°F for lower shell plate B6903-1 will be reached at a fluence level of 4.961E+19 n/cm^2 (E>1.0 MeV), which is equivalent to 43.87 EFPY. The Unit 1 Reactor Vessel is projected to reach the PTS screening criterion of 270°F on the limiting plate (B6903-1) in the year 2033.

10 CFR 50.61 [Reference A.2-7] allows that:

> *For each pressurized water nuclear power reactor for which the value of RT_{PTS} for any material in the beltline is projected to exceed the PTS screening criterion using the EOL fluence, the licensee shall implement those flux reduction programs that are reasonably practicable to avoid exceeding the PTS screening criterion set forth in paragraph (b)(2) of this section.*

Therefore, a sensitivity assessment of available flux reduction measures was completed. The sensitivity assessment included several fuel management scenarios (such as low leakage core design, low power peripheral fuel assemblies, reinsertion of hafnium rods, and the use of part length shielded assemblies) and several assumed capacity factors up to 98 percent. Several flux reduction options are available which would maintain the limiting plate below the PTS screening criterion to the EOLE. The flux reduction program will be managed under the Reactor Vessel Integrity Program. Documentation of a flux reduction program for Unit 1 will be submitted in accordance with the requirements of 10 CFR 50.61.

The Unit 1 Reactor Vessel fluence will continue to be monitored as part of the Reactor Vessel Integrity Program to ensure the projected fluence remains below that assumed for the relevant neutron embrittlement TLAA. Therefore, the Unit 1 RT_{PTS} TLAA will be adequately managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

A.2.2.3 Charpy Upper Shelf Energy

In the spring of 2000, Surveillance Capsule Y was pulled and the analysis was documented in WCAP-15571 [Reference A.2-4]. For license renewal, WCAP-15571 Supplement 1 [Reference A.2-5] documents the EOLE analysis for Charpy upper-shelf energy (C_VUSE).

For Unit 1, there exists material surveillance data for Reactor Vessel lower shell plate B6903-1 (heat C6317-1) and the intermediate shell longitudinal weld (heat 305424). The measured drops in C_VUSE for each of these material heats was plotted on Figure 2 of Regulatory Guide 1.99, *Radiation Embrittlement of Reactor Vessel Materials* [Reference A.2-8], with a horizontal line drawn parallel to the existing lines as the upper bound of all data. Regulatory Guide 1.99, Figures 1 and 2, were used in the determination of the percent decrease in C_VUSE for the beltline and extended beltline materials.

The beltline and extended beltline material C_VUSE values were determined to maintain 50 ft-lb or greater at 54 EFPY. Therefore, the Unit 1 C_VUSE analysis has been dispositioned in accordance with 10 CFR 54.21(c)(1)(ii).

A.2.2.4 Pressure-Temperature Limits

BVPS pressure-temperature (P-T) limit curves are operating limits, conditions of the operating license, and are included in the Pressure and Temperature Limits Report, as required by Technical Specifications. They are valid up to a stated vessel fluence limit, and must be revised prior to operating beyond that limit. The provisions of 10 CFR 50, Appendix G [Reference A.2-7], require BVPS to operate within the currently licensed P-T limit curves. These curves are required to be maintained and updated as necessary to maintain plant operation consistent with 10 CFR 50. The Reactor Vessel Integrity Program will maintain the P-T limit curves for Unit 1 for the period of extended operation. Therefore, the Unit 1 P-T limit curves TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

At BVPS, the Low-Temperature Overpressure Protection System is known as the Overpressure Protection System (OPPS). As part of any update, the OPPS setpoints (OPPS enable temperature and power-operated relief valve setpoints) for both units are reviewed and updated as required based on the updated P-T limit curves.

A.2.3 METAL FATIGUE

The analysis of metal fatigue is a TLAA for Class 1 and selected non-Class 1 mechanical components within the scope of license renewal. The following sections summarize the analyses associated with metal fatigue of fluid systems:

- Class 1 Fatigue Evaluations (Section A.2.3.1)
- Non-Class 1 Fatigue Evaluations (Section A.2.3.2)
- Generic Industry Issues on Fatigue (Section A.2.3.3)

A.2.3.1 Class 1 Fatigue Evaluations

The design of BVPS Class 1 components incorporates the requirements of Section III of the ASME Code, which requires a discrete analysis of the thermal and dynamic stress cycles on components that make up the reactor coolant pressure boundary. The fatigue analyses rely on the definition of design basis transients that envelope the expected cyclic service and the calculation of a cumulative usage factor (CUF). In accordance with ASME Section III, Subsection NB, the cumulative usage factor shall not exceed 1.0. The required analysis was performed for BVPS, and incorporated a set of design basis transients based on the original 40-year operating life of the plant. These ASME Section III, Class 1 fatigue evaluations are contained in the specific piping and component analyses and stress reports and, because they are based on a number of design cycles assumed for the life of the plant, these evaluations are TLAAs.

The BVPS original design basis transients including design cycles for the RCS are identified in Table 4.1-10 of the UFSAR. BVPS reviewed the design cycles against 60-year projected operational cycles and determined that the design cycles are bounding for the period of extended operation. Since the 60-year projected operational cycles were used in determining that the design fatigue analyses remain valid for 60 years, the Metal Fatigue of Reactor Coolant Pressure Boundary Program must continue to be used to validate the assumptions used in the evaluations. Therefore, Class 1 components and piping fatigue TLAAs have been dispositioned in accordance with 10 CFR 54.21(c)(1)(i) and 10 CFR 54.21(c)(1)(iii).

A.2.3.1.1 Unit 1 Pressurizer

In 1999, the analysis of the Unit 1 pressurizer, lower shell and related components was revised to address improvements to the insurge/outsurge transients identified by the Westinghouse Owners Group. Plant operating procedures were revised to follow the guidance of the Westinghouse Owners Group and to minimize the impact of potential insurges. Prior to the 1999 reanalysis, BVPS Unit 1 had experienced several pressurizer spray transients that challenged the analytical and Technical Specification limit of 320°F difference between the spray line temperature and the pressurizer steam space temperature. Revised transients for initial spray flow were incorporated into the analysis. In 2005, BVPS decided to further revise the operating procedures to optimize the plant shutdown and startup processes. The BVPS Optimized procedures have been shown to meet all recommendations of the Westinghouse Owners Group and have virtually eliminated the potential for insurges. Next, the Extended Power Uprate Project evaluated the revised Uprate transients against the previous analysis. The cumulative usage factors associated with the Unit 1 pressurizer are less than 1.0. Since the 60-year projected operational cycles were used in determining that the pressurizer design fatigue analysis remains valid for 60 years, the Metal Fatigue of Reactor Coolant Pressure Boundary Program must continue to be used to validate the assumptions used in the evaluation. In addition, the pressurizer insurge cycle assumptions used in the pressurizer analysis require validation for the period of extended operation. The Metal Fatigue of Reactor Coolant Pressure Boundary Program identifies the pressurizer insurge transient as a supplemental transient that requires monitoring. Therefore, the pressurizer fatigue TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

A.2.3.2 Non-Class 1 Fatigue Evaluations

A.2.3.2.1 Piping and In-Line Components

The design code for non-Class 1 piping and in-line components (e.g., fittings and valves) within the scope of license renewal is ANSI B31.1 or ASME III Subsections NC and ND. These codes specify evaluation of cyclic secondary stresses (i.e., stresses due to thermal expansion and anchor movements) by applying stress range reduction factors against the allowable stress range (S_A).

For all those non-Class 1 components identified as subject to cracking due to fatigue, a review of system operating characteristics was conducted by BVPS to determine the approximate frequency of any significant thermal cycling. If the number of equivalent full temperature cycles is below the limit used for the current design, the component is suitable for extended operation. If the number of equivalent full temperature cycles exceeds the limit, evaluation of the individual stress calculations will be required.

BVPS evaluated the validity of this assumption for 60 years of plant operation. The results of this evaluation indicate that the thermal cycle assumption is valid and bounding for 60 years of operation. Therefore, these piping fatigue analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.2.3.3 Generic Industry Issues on Fatigue

This section addresses the BVPS fatigue TLAAs associated with NRC Bulletins 88-08 and 88-11. In addition, this section addresses the effects of the primary coolant environment on fatigue life.

A.2.3.3.1 Pressurizer Surge Line Thermal Stratification (NRC Bulletin 88-11)

NRC Bulletin 88-11, *Pressurizer Surge Line Thermal Stratification* [Reference A.2-9], required a plant-specific or generic analysis demonstrating that the pressurizer surge line meets the applicable design code requirements considering the effects of thermal stratification.

In response to the Bulletin, BVPS submitted a plant-specific analysis, WCAP-12727, *Evaluation of Thermal Stratification for the Beaver Valley Unit 1 Pressurizer Surge Line* [Reference A.2-10], to the NRC. The NRC approved [Reference A.2-11] this evaluation. WCAP-12727 determined the effects of thermal stratification in the surge line through the imposition of defined thermal stratification cycles upon the stress and fatigue evaluations. The stratification cycles incorporated into the cumulative usage factor determination are defined by the 200 heatup and cooldown design transients. Therefore, this NRC Bulletin 88-11 analysis is a TLAA in accordance with 10 CFR 54.3.

WCAP-12727 was reviewed for impact due to extended power uprate. A detailed analysis was performed at the controlling location (reactor coolant loop nozzle) to account for temperature effects due to the power uprate. A new cumulative usage factor was calculated and demonstrated to remain less than the Code allowable limit of 1.0.

The 200 heatup and cooldown transients were determined to remain bounding for the period of extended operation. Therefore, the Unit 1 pressurizer surge line fatigue TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

A.2.3.3.2 Effects of Primary Coolant Environment on Fatigue Life

Test data indicate that certain environmental conditions (such as temperature, oxygen content, and strain rate) in the primary systems of light water reactors could result in greater susceptibility to fatigue than would be predicted by fatigue analyses based on the ASME Section III design fatigue curves. One NRC study, documented in NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components* [Reference A.2-12], applied the fatigue design curves that incorporated environmental effects to several plant designs. The results of studies performed on this topic, including NUREG/CR-6260, were summarized in Generic Safety Issue (GSI)-190, *Fatigue*

Evaluation of Metal Components for 60-Year Plant Life [Reference A.2-13]. In closing GSI-190, regarding the effects of a reactor water environment on fatigue life, the NRC concluded that licensees should address the effects of the coolant environment on component fatigue life as aging management programs are formulated in support of license renewal.

The Unit 1 reactor coolant pressure boundary piping is designed to B31.1, and is therefore classified as an older-vintage Westinghouse plant.

Section 5.5 of NUREG/CR-6260 identified the following component locations as representative for environmental effects for older-vintage Westinghouse plants. These locations and the subsequent calculations are directly relevant to Unit 1, and include the:

- Reactor vessel shell and lower head (shell-to-head transition);
- Reactor vessel inlet and outlet nozzles;
- Pressurizer surge line (hot leg nozzle safe end);
- RCS piping charging system nozzle;
- RCS piping safety injection nozzle; and,
- RHR system tee.

The NUREG/CR-6260 locations were evaluated using the guidance of NUREG/CR-6583, *Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low Alloy Steels* [Reference A.2-14], and NUREG/CR-5704, *Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels* [Reference A.2-15]. These reports describe the use of a fatigue life correction factor (F_{en}) to express the effects of the reactor coolant environment upon the material fatigue life. The expression for F_{en} was determined through experimental and statistical data. F_{en} for carbon and low alloy steel is a function of fluid service temperature, material sulfur content, fluid dissolved oxygen, and strain rate. For austenitic stainless steel, F_{en} is a function of fluid service temperature, fluid dissolved oxygen, and strain rate. The cumulative usage factor which includes environmental effects (U_{env}) is determined from the existing 60-year cumulative usage factor (U_{60}) through the use of the fatigue life correction factor:

$$U_{env} = U_{60} * F_{en}$$

To demonstrate acceptable fatigue life including environmental effects, the cumulative usage factor, which includes environmental effects, should remain less than design code allowables (i.e., $U_{env} \leq 1.0$). Therefore, F_{en} was applied to the cumulative usage factors at the NUREG/CR-6260 locations and compared to the design code allowable limit.

At two locations (pressurizer surge line and charging system nozzle), U_{env} exceeded the design code allowable limit of 1.0. For these locations, BVPS will implement one or more of the following as required by the Metal Fatigue of Reactor Coolant Pressure Boundary Program:

1. Further refinement of the fatigue analyses to lower the predicted CUFs to less than 1.0;
2. Management of fatigue at the affected locations by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method acceptable to the NRC); or,
3. Repair or replacement of the affected locations.

Should BVPS select the option to manage environmentally-assisted fatigue during the period of extended operation, details of the aging management program, such as scope, qualification, method, and frequency, will be submitted to the NRC prior to the period of extended operation. Therefore, the pressurizer surge line and charging system nozzle TLAAs have been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

The cumulative usage factors including environmental fatigue at the other locations (reactor vessel shell and lower head, reactor vessel inlet and outlet nozzles, safety injection nozzle and RHR system tee) have been demonstrated to remain less than the design code allowable limit of 1.0 for the period of extended operation. Therefore, the fatigue TLAAs associated with these locations have been dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

A.2.4 ENVIRONMENTAL QUALIFICATION (EQ) OF ELECTRIC EQUIPMENT

The BVPS existing Environmental Qualification (EQ) of Electric Components Program manages component thermal, radiation and cyclical aging, as applicable, through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods. As required by 10 CFR 50.49, environmental qualification components not qualified for the current license term are to be refurbished, replaced, or have their qualification extended prior to reaching the aging limits established in the evaluation. The Environmental Qualification of Electric Components Program ensures that these environmental qualification components are maintained in accordance with their qualification bases. Aging evaluations for environmental qualification components that specify a qualification of at least 40 years are time-limited aging analyses for license renewal.

The Environmental Qualification (EQ) of Electric Components Program is an existing program established to meet BVPS commitments for 10 CFR 50.49. Continued implementation of the Environmental Qualification (EQ) of Electrical Components Program provides reasonable assurance that the aging effects will be managed and that the in-scope EQ components will continue to perform their intended function(s) for the period of extended operation. The effects of aging will be managed by the program in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

A.2.5 CONTAINMENT LINER PLATE, METAL CONTAINMENT, AND PENETRATIONS FATIGUE

Several potential TLAA associated with the Containment structure were identified and are summarized in the following sections:

- Containment Liner Fatigue (Section A.2.5.1)
- Containment Liner Corrosion Allowance (Section A.2.5.2)
- Containment Liner Penetration Fatigue (Section A.2.5.3)

A.2.5.1 Containment Liner Fatigue

The Unit 1 Containment liner stress analysis determines a fatigue usage factor based on specific design cyclic loads in accordance with paragraph N-415.2 of the 1968 Edition of ASME Section III. These design loads include 1000 cycles of pressure variation due to normal operations (startup and shutdown), 4000 cycles of temperature variation due to normal operations (startup and shutdown), and 20 cycles of design basis earthquake. The usage factor for the liner was determined to be significantly less than 1.0. The anticipated occurrences of these cycles are described in Table 5.2-13 of the Unit 1 UFSAR as follows:

- 150 cycles of loading due to the differential pressure between operating and atmospheric pressure are anticipated on the basis of 2.5 refueling cycles per year on a 60-year span;
- 600 cycles of loading due to thermal expansion resulting when the liner is exposed to the differential temperature between operating and seasonal refueling temperatures are anticipated on the basis of 10 such variations per year on a 60-year span; and,
- 150 cycles of operating basis earthquake, which is an assumed number of cycles of this type of earthquake for a 60-year span.

As shown above, the design cycles of the Unit 1 Containment liner bound the anticipated pressure and temperature cycles expected through the period of extended operation. The expected stresses resulting from the 60-year anticipated operating basis earthquake cycles were determined to be bounded by those due to the analyzed design basis earthquake cycles. Therefore, the Unit 1 Containment liner fatigue TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(ii).

A.2.5.2 Containment Liner Corrosion Allowance

The Reactor Containment Building has a continuously welded carbon steel liner which acts as a leak-tight membrane. The cylindrical portion of the liner is 3/8-inch thick, the hemispherical dome is 1/2-inch thick, and the flat floor liner covering the concrete mat is 1/4-inch thick. The floor liner plate is covered with approximately two feet of reinforced concrete. All welded seams were originally covered with continuously welded leak test channels that were installed to facilitate leak testing of welds during liner erection. Since initial construction, several test channels have been removed. Also, test channels were not installed on liner plate seams associated with the Unit 1 Steam Generator Replacement Project construction opening. Channels in the hemispherical dome and Containment mat are covered with concrete while those on the cylindrical liner wall are exposed. Test ports that were provided for leak testing were sealed with vent plugs after the completion of the testing. These plugs were to remain in place during subsequent Type-A leak rate testing.

During a Unit 1 shutdown in 1991, it was determined that 27 vent plugs in the Containment floor liner test channels were missing. The missing test channel vent plugs allowed moisture and condensation inside the test channels, leading to minor corrosion of the liner. BVPS evaluated the test channels to determine the impact to the Containment liner, and submitted the results of the evaluations to the NRC as Amendments 165 and 47, Unit 1 and Unit 2 respectively, to the operating licenses. These amendments were approved by the NRC and documented in an SER [Reference A.2-16]. After further evaluation, it was concluded that these initial evaluations contained some nonconservative assumptions with regard to the corrosion rates in the test channels. BVPS took corrective action to arrest the corrosion rate in the affected test channels, including inerting and sealing the test channels. The further evaluation and corrective actions are documented in a 1992 Letter to the NRC [Reference A.2-17]. These corrosion rate analyses meet the 10 CFR 54.3 requirements as TLAAs and must be evaluated for the period of extended operation.

The minimum required thickness for the Containment liner has been determined for the various portions of the liner. The limiting liner portion is the liner floor plate, which has a fabrication thickness of 0.25 inches and a minimum required thickness of 0.125 inches. Thus, the corrosion allowance is 0.125 inches (125 mils). The inerting and sealing of the test channels significantly reduced the theoretical corrosion rates in the channels. The total estimated penetration due to corrosion of the inerted channel was estimated at 69.2 mils for 43 years of plant operation. The maximum expected corrosion rate for the carbon steel liner in this low oxygen environment was determined to be 0.39 mils per year. Therefore, projecting the expected corrosion penetration with the maximum expected corrosion rate to the end of the period of extended operation results in an additional 7.8 mils of corrosion. Adding this to the previous expected corrosion penetration depths yields 77.0 mils of corrosion penetration. This result is well within the corrosion allowance of 125 mils.

Therefore, the Unit 1 Containment liner corrosion analysis has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A.2.5.3 Containment Liner Penetration Fatigue

A.2.5.3.1 Equipment Hatch

The equipment hatch and integral emergency airlock are designed and analyzed in accordance with ASME Section III, Division 1, Subsection NE (Class MC). Subsection NE states that any portions not satisfying the fatigue exemption as described in Subsection NB-3222(d) require further fatigue evaluation. Therefore, a fatigue exemption was completed for the Unit 1 equipment hatch in accordance with Subsection NB-3222(d). This exemption was based on assumed cycles for a 40-year life, namely 10 pressurization events due to LOCA, and 80 cycles of startup and shutdown. It is highly unlikely that Unit 1 will reach 10 pressurization events due to LOCA for 60 years of operation. The assumption of 80 cycles of startup and shutdown is not bounding for 60 years of operation. A reanalysis was performed using 240 startup and shutdown cycles that bounds the number of projected cycles for the period of extended operation. Therefore, the equipment hatch fatigue TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(ii).

A.2.5.3.2 Fuel Transfer Tube

The fuel transfer tube pipe was analyzed to ASME Section III, Division 1, Subsection NC. The analysis for the fuel transfer tube pipe uses a stress range reduction factor of 1.0 (<7,000 cycles). However, as the fuel transfer tube pipe experiences operational cycles only during refueling, the fuel transfer tube pipe experiences essentially no thermal cycles. The existing fuel transfer tube pipe fatigue TLAA remains valid through the period of extended operation. Therefore, the Unit 1 fuel transfer tube pipe fatigue TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

The fuel transfer tube bellows were analyzed to ASME Section III, Division 1, Subsection NC. The bellows stress analyses determined acceptability based on the bellows experiencing displacements due to a design basis earthquake. The assumed design cycles were 600. This number of design basis earthquake cycles is highly unlikely to occur during the period of extended operation. The fuel transfer tube bellows fatigue TLAAs remain valid through the period of extended operation. Therefore, the fuel transfer tube bellows fatigue TLAAs have been dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

A.2.5.3.3 Containment Penetration Bellows

The bellows (metal expansion joints) are part of the system evaluation boundary of the River Water System and are located at the discharge piping connections from the recirculation spray heat exchangers inside Containment. The piping and in-line components of the River Water System are designed and analyzed to the 1967 Edition of B31.1. This code specifies evaluation of cyclic secondary stresses (i.e., stresses due to thermal expansion and anchor movements) by applying stress range reduction factors against the allowable stress range (SA).

For those non-Class 1 components identified as subject to cracking due to fatigue, a review of system operating characteristics was conducted to determine the approximate frequency of significant thermal cycling. If the number of equivalent full-temperature cycles is below the limit used for the current design (7,000 cycles in this case), the component is suitable for extended operation. If the number of equivalent full-temperature cycles exceeds the limit, evaluation of the individual stress calculations will be required.

BVPS evaluated the validity of this assumption for 60 years of plant operation. The Recirculation Spray System is normally in standby operation, and, including any periodic testing, will experience significantly less than the full-temperature cycle limit of 7,000 cycles for the period of extended operation. Therefore, the Recirculation Spray System fatigue analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.2.6 OTHER PLANT-SPECIFIC TIME-LIMITED AGING ANALYSES

The plant-specific TLAAs summarized in this section include:

- Piping Subsurface Indications (Section A.2.6.1)
- Reactor Vessel Underclad Cracking (Section A.2.6.2)
- Leak Before Break (Section A.2.6.3)
- Crane Load Cycles (Section A.2.6.4)

A.2.6.1 Piping Subsurface Indications

During a Unit 1 inservice inspection performed in the Cycle 11 Refueling Outage (March - May, 1996), an indication was identified on the RCS loop C cold leg between an elbow and a section of straight pipe which exceeded the ASME Code, Section XI, subsection IWB-3500 acceptance criteria. This section of pipe is Class 1 cast austenitic stainless steel (CASS) piping. Subsequently, an analysis was performed to ensure that this indication would remain within ASME Code, Section XI, Appendix C evaluation acceptance standards. This evaluation, approved by the NRC [Reference A.2-18], concluded that the postulated flaw met the applicable requirements with significant margins of safety to the end of the service lifetime. This flaw growth evaluation is a TLAA because it contained two parameters that are based on the service life of the piping, namely thermal aging and fatigue transient cycles.

Thermal aging in CASS will continue until the saturation, or fully-aged, point is reached. The limiting fracture toughness properties were those of the straight pipe, which has a relatively high ferrite content. Therefore, the fully aged (saturated) fracture toughness properties of the straight pipe were used in the analysis. Since the analysis relies on fully aged stainless steel material properties, the analysis does not have a material property time-dependency that requires further evaluation for license renewal.

The flaw evaluation includes the postulation of an initial flaw and the growth of that flaw based on imposed loading transients. The cycle assumptions used in the analysis are conservative compared to the BVPS original design transients. The BVPS original design basis transients including design cycles are identified in Table 4.1-10 of the UFSAR. BVPS has reviewed the design cycles against 60-year projected operational cycles and has determined that the design cycles are bounding for the period of extended operation. Since the 60-year projected operational cycles were used in determining that the flaw growth analysis remains valid for 60 years, the Metal Fatigue of Reactor Coolant Pressure Boundary Program must continue to be used to validate the assumptions used in the evaluation. Therefore, the Unit 1 flaw growth TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(i) and 10 CFR 54.21(c)(1)(iii).

A.2.6.2 Reactor Vessel Underclad Cracking

WCAP-15338-A, *A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants* [Reference A.2-20], evaluates the impact of cracks beneath austenitic stainless steel weld cladding on reactor pressure vessel integrity for 60 years of operation.

The Unit 1 Reactor Vessel does not contain SA 508, Class 2 forgings in the beltline regions. Only the vessel and closure head flanges and the inlet and outlet nozzles are fabricated from SA 508, Class 2 forgings. The evaluation contained in WCAP-15338-A has been used to demonstrate that fatigue growth of the subject flaws will be minimal over 60 years and the presence of the underclad cracks are of no concern relative to the structural integrity of the Reactor Vessel.

The cycle assumptions used in the flaw growth analysis are conservative compared to the BVPS original design transients. The BVPS original design basis transients including design cycles are identified in Table 4.1-10 of the UFSAR. BVPS has reviewed the design cycles against 60-year projected operational cycles and has determined that the design cycles are bounding for the period of extended operation. Since the 60-year projected operational cycles were used in determining that the flaw growth analysis remains valid for 60 years, the Metal Fatigue of Reactor Coolant Pressure Boundary Program must continue to be used to validate the assumptions used in the evaluation. Therefore, the Unit 1 flaw growth TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(i) and 10 CFR 54.21(c)(1)(iii).

A.2.6.3 Leak Before Break

Leak before break (LBB) analyses evaluate postulated flaw growth in piping to alter the structural design basis. BVPS has determined that the fatigue crack growth analysis is a TLAA that requires disposition for license renewal.

For the LBB analyses discussed in the following two subsections, the only consideration that could be influenced by time is the accumulation of actual fatigue transient cycles. The cycle assumptions used in the analyses are conservative compared to the BVPS original design transients. The BVPS original design basis transients including design cycles are identified in Table 4.1-10 of the UFSAR. BVPS has reviewed the design cycles against 60-year projected operational cycles and has determined that the design cycles are bounding for the period of extended operation. Since the 60-year projected operational cycles were used in determining that the flaw growth analyses remain valid for 60 years, the Metal Fatigue of Reactor Coolant Pressure Boundary Program must continue to be used to validate the assumptions used in the evaluations. Therefore, the Unit 1 flaw growth TLAAs have been dispositioned in accordance with 10 CFR 54.21(c)(1)(i) and 10 CFR 54.21(c)(1)(iii).

A.2.6.3.1 Main Coolant Loop Piping Leak Before Break

The current LBB evaluation for the main coolant loop piping is documented in WCAP-11317, *Technical Justification for Eliminating Large Primary Loop Pipe*

Rupture as the Structural Design Basis for Beaver Valley Unit 1 [Reference A.2-21]. This evaluation (including Supplements 1 and 2) was approved by the NRC in a Safety Evaluation Report [Reference A.2-22] in 1987.

A.2.6.3.2 Pressurizer Surge Line Piping Leak Before Break

The current LBB evaluation for the pressurizer surge line piping is documented in WCAP-12727, *Evaluation of Thermal Stratification for the Beaver Valley Unit 1 Pressurizer Surge Line* [Reference A.2-23]. This evaluation was approved by the NRC in a Safety Evaluation Report [Reference A.2-24] in 1991.

A.2.6.4 Crane Load Cycles

In the response to NUREG-0612, *Control of Heavy Loads at Nuclear Power Plants* [Reference A.2-26], BVPS determined that two cranes, the fuel cask crane (CR-15), and the moveable platform and hoists (CR-27), were designed to comply with Crane Manufacturers Association of America Specification #70 (CMAA-70), *Specifications for Electric Overhead Traveling Cranes* [Reference A.2-27]. Therefore, these cranes have a TLAA associated with their design calculations.

These cranes may conservatively be classified as Service Class A cranes. The total load cycles and mean effective load factors for the cranes have been estimated for the period of extended operation. Even using conservative estimates, total load cycles are well below 20,000, and mean effective load factors are maintained within or below the Service Class A bounds (0.35 - 0.53) for 60 years. Therefore, crane allowable stress ranges as defined in CMAA-70 will remain valid through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.2.7 APPENDIX A.2 REFERENCES

A.2-1 [BVPS License Renewal Application – later].

A.2-2 [NRC SER for BVPS License Renewal – later].

A.2-3 10 CFR 54, *Requirements for Renewal of Operating Licenses for Nuclear Power Plants*.

A.2-4 WCAP-15571, *Analysis of Capsule Y from First Energy Company Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program*, Rev. 0.

A.2-5 WCAP-15571 Supplement 1, *Analysis of Capsule Y from First Energy Company Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program*, June 2007.

A.2-6 Regulatory Guide 1.190, *Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence*, March 2001.

A.2-7 10 CFR 50, *Domestic Licensing of Production and Utilization Facilities*.

A.2-8 Regulatory Guide 1.99, *Radiation Embrittlement of Reactor Vessel Materials*, Rev. 2.

A.2-9 NRC Bulletin 88-11, *Pressurizer Surge Line Thermal Stratification*, December 20, 1988.

A.2-10 WCAP-12727, *Evaluation of Thermal Stratification for the Beaver Valley Unit 1 Pressurizer Surge Line*, Rev. 0.

A.2-11 De Agazio, Albert W. (NRC), Letter to John D. Sieber (BVPS), *Approval of Leak-Before-Break Analysis (TAC No. 72110)*, May 2, 1991.

A.2-12 NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components*, February 1995.

A.2-13 Generic Safety Issue (GSI)-190, *Fatigue Evaluation of Metal Components for 60-Year Plant Life*, Rev. 2.

A.2-14 NUREG/CR-6583, *Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low Alloy Steels*, February 1998.

A.2-15 NUREG/CR-5704, *Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels*, March 1999.

A.2-16 De Agazio, Albert W. (NRC), Letter to J. D. Sieber (BVPS), *Beaver Valley Units 1 And 2 - Issuance of Amendments 165 and 47: Containment Structural Integrity - Change Request Nos. 181/45*, June 23, 1992.

A.2-17 Sieber, J. D. (BVPS), Letter to NRC, *Beaver Valley Power Station, Unit No. 1 and No. 2, BV-1 Docket No. 50-334, License No. DPR-66, BV-2 Docket No. 50-412, License No. NPF-73, Revision to SER for Amendments 165 and 47*, December 30, 1992.

A.2-18 Brinkman, Donald S. (NRC), Letter to J.E. Cross (BVPS), *Evaluation of Flaw Indication in Reactor Coolant System (RCS) Cold Leg Pipe Weld, Beaver Valley Power Station, Unit No. 1 (BVPS-1)*, May 1, 1996.

A.2-19 WCAP-7733, *Reactor Vessels Weld Cladding - Base Metal Interaction*, July 1971.

A.2-20 WCAP-15338-A, *A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants*, October 2002.

A.2-21 WCAP-11317, *Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Beaver Valley Unit 1*, March 1987 (including Supplements 1 and 2).

A.2-22 Tam, Peter S. (NRC), Letter to J.D. Sieber (BVPS), *Beaver Valley Unit 1 - Removal of Large-Bore Snubbers from Primary Coolant Loops*, December 9, 1987.

A.2-23 WCAP-12727, *Evaluation of Thermal Stratification for the Beaver Valley Unit 1 Pressurizer Surge Line*, Rev. 0.

A.2-24 De Agazio, Albert W. (NRC), Letter to J. D. Sieber (BVPS), *Approval of Leak-Before-Break Analysis*, May 2, 1991.

A.2-25 Regulatory Guide 1.46, *Protection Against Pipe Whip Inside Containment*, May 1973.

A.2-26 NUREG-0612, *Control of Heavy Loads at Nuclear Power Plants*, July 1980.

A.2-27 Crane Manufacturers Association of America Specification #70 (CMAA-70), *Specifications for Electric Overhead Traveling Cranes*, Revised 1983.

A.3 EVALUATION SUMMARIES OF UNIT 2 TIME-LIMITED AGING ANALYSES

A.3.1 INTRODUCTION

Time-limited aging analyses (TLAAs) are defined in 10 CFR 54.3 [Reference A.3-3] as:

> *Time-limited aging analyses, for the purposes of this part, are those licensee calculations and analyses that:*
>
> 1. *Involve systems, structures, and components within the scope of license renewal, as delineated in §54.4(a);*
> 2. *Consider the effects of aging;*
> 3. *Involve time-limited assumptions defined by the current operating term, for example, 40 years;*
> 4. *Were determined to be relevant by the licensee in making a safety determination;*
> 5. *Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, and component to perform its intended functions, as delineated in §54.4(b); and*
> 6. *Are contained or incorporated by reference in the CLB.*

Once identified, TLAAs must be evaluated and dispositioned as described in the following section of 10 CFR 54:

> §54.21 *Contents of application -- technical information.*
>
> *(c) An evaluation of time-limited aging analyses.*
>
> 1. *A list of time-limited aging analyses, as defined in §54.3, must be provided. The applicant shall demonstrate that —*
>   - *(i). The analyses remain valid for the period of extended operation;*
>   - *(ii). The analyses have been projected to the end of the period of extended operation; or*
>   - *(iii). The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.*

This chapter provides a summary of the TLAAs identified in the BVPS License Renewal Application, and includes the following topics:

- Reactor Vessel Neutron Embrittlement (Section A.3.2)

- Metal Fatigue (Section A.3.3)
- Environmental Qualification (EQ) of Electric Equipment (Section A.3.4)
- Containment Liner Plate, Metal Containment, and Penetrations Fatigue (Section A.3.5)
- Other Plant-Specific Time-Limited Aging Analyses (Section A.3.6)
- Appendix A.3 References (Section A.3.7)

A.3.2 REACTOR VESSEL NEUTRON EMBRITTLEMENT

Four analyses that address the effects of neutron irradiation embrittlement of the Reactor Vessel have been identified as TLAAs. These analyses are summarized in the following sections:

- Neutron Fluence Values (Section A.3.2.1)
- Pressurized Thermal Shock (Section A.3.2.2)
- Charpy Upper Shelf Energy (Section A.3.2.3)
- Pressure-Temperature Limits (Section A.3.2.4)

A.3.2.1 Neutron Fluence Values

Loss of fracture toughness is an aging effect caused by the neutron embrittlement aging mechanism that results from prolonged exposure to neutron radiation. This process results in increased tensile strength and hardness of the material with reduced toughness. The rate of neutron exposure is defined as neutron flux, and the cumulative degree of exposure over time is defined as neutron fluence. As neutron embrittlement progresses, the toughness/temperature curve shifts downward (lower fracture toughness), and the curve shifts to the right (brittle/ductile transition temperature increases).

In the spring of 2005, Surveillance Capsule X was pulled and the analysis was documented in WCAP-16527-NP, *Analysis of Capsule X from First Energy Nuclear Operating Company Beaver Valley Unit 2 Reactor Vessel Radiation Surveillance Program* [Reference A.3-4]. For license renewal, WCAP-16527-NP Supplement 1, *Analysis of Capsule X from First Energy Company Beaver Valley Unit 2 Reactor Vessel Radiation Surveillance Program* [Reference A.3-5] documents the end-of-license-extended (EOLE) analysis for neutron fluence values.

The fluence values were projected using ENDF/B-VI cross sections, are based on the results of the Capsule X analysis, and comply with Reg. Guide 1.190, *Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence* [Reference A.3-6].

The fluence projections include fuel cycle-specific calculated neutron exposures at the end of Cycle 11 (April 2005) as well as future projections for several intervals extending to 54 EFPY. The projections were based on the assumption that the core power distributions and associated plant operating characteristics for Cycle 12 were representative of plant operation to 17 EFPY and that the preliminary Cycle 13 (began November 2006) core power distributions were applicable beyond 17 EFPY. The calculations account for a core power uprate from 2689 MWt to 2900 MWt at 17 EFPY.

A.3.2.2 Pressurized Thermal Shock

In the spring of 2005, Surveillance Capsule X was pulled and the analysis was documented in WCAP-16527-NP [Reference A.3-4]. For license renewal, WCAP-16527-NP Supplement 1 [Reference A.3-5] documents the EOLE analysis for pressurized thermal shock (PTS).

Using the prescribed PTS Rule (10 CFR 50.61 [Reference A.3-7]) methodology, reference temperature for pressurized thermal shock (RT_{PTS}) values were generated for beltline and extended beltline region materials of the BVPS Unit 2 Reactor Vessel for fluence values at EOLE (54 EFPY). The projected RT_{PTS} values for EOLE (54 EFPY) meet the 10 CFR 50.61 screening criteria for beltline and extended beltline materials. Therefore, the Unit 2 RT_{PTS} TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(ii).

A.3.2.3 Charpy Upper Shelf Energy

In the spring of 2005, Surveillance Capsule X was pulled and the analysis was documented in WCAP-16527-NP [Reference A.3-4]. For license renewal, WCAP-16527-NP Supplement 1 [Reference A.3-5] documents the EOLE analysis for Charpy upper-shelf energy (C_VUSE).

For Unit 2, there exists material surveillance data for Reactor Vessel intermediate shell plate B9004-2 (heat C0544-2) and the intermediate shell longitudinal weld (heat 83642). The measured drops in C_VUSE for each of these material heats was plotted on Figure 2 of Regulatory Guide 1.99, *Radiation Embrittlement of Reactor Vessel Materials* [Reference A.3-8], with a horizontal line drawn parallel to the existing lines as the upper bound of all data. Regulatory Guide 1.99 Figures 1 and 2 were used in the determination of the percent decrease in C_VUSE for the beltline and extended beltline materials.

The beltline and extended beltline material C_VUSE values were determined to maintain 50 ft-lb or greater at 54 EFPY. Therefore, the Unit 2 C_VUSE analysis has been dispositioned in accordance with 10 CFR 54.21(c)(1)(ii).

A.3.2.4 Pressure-Temperature Limits

BVPS pressure-temperature (P-T) limit curves are operating limits, conditions of the operating license, and are included in the Pressure and Temperature Limits Report, as required by Technical Specifications. They are valid up to a stated vessel fluence limit, and must be revised prior to operating beyond that limit. The provisions of 10 CFR 50, Appendix G [Reference A.3-7], require BVPS to operate within the currently licensed P-T limit curves. These curves are required to be maintained and updated as necessary to maintain plant operation consistent with 10 CFR 50. The Reactor Vessel Integrity Program will maintain the P-T limit curves for Unit 2 for the period of extended operation. Therefore, the Unit 2 P-T limit curves TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

At BVPS, the Low-Temperature Overpressure Protection System is known as the Overpressure Protection System (OPPS). As part of any update, the OPPS setpoints (OPPS enable temperature and power-operated relief valve setpoints) for both units are reviewed and updated as required based on the updated P-T limit curves.

A.3.3 METAL FATIGUE

The analysis of metal fatigue is a TLAA for Class 1 and selected non-Class 1 mechanical components within the scope of license renewal. The following sections summarize the analyses associated with metal fatigue of fluid systems:

- Class 1 Fatigue Evaluations (Section A.3.3.1)
- Non-Class 1 Fatigue Evaluations (Section A.3.3.2)
- Generic Industry Issues on Fatigue (Section A.3.3.3)

A.3.3.1 Class 1 Fatigue Evaluations

The design of BVPS Class 1 components incorporates the requirements of Section III of the ASME Code, which requires a discrete analysis of the thermal and dynamic stress cycles on components that make up the reactor coolant pressure boundary. The fatigue analyses rely on the definition of design basis transients that envelope the expected cyclic service and the calculation of a cumulative usage factor (CUF). In accordance with ASME Section III, Subsection NB, the CUF shall not exceed 1.0. The required analysis was performed for BVPS and incorporated a set of design basis transients based on the original 40-year operating life of the plant. These ASME Section III, Class 1 fatigue evaluations are contained in the specific piping and component analyses and stress reports and, because they are based on a number of design transient cycles assumed for the life of the plant, these evaluations are TLAAs.

The BVPS original design basis transients including design cycles for the RCS are identified in Table 3.9N-1 of the UFSAR. BVPS has reviewed the design cycles against 60-year projected operational cycles and has determined that the design cycles are bounding for the period of extended operation, except in certain specific cases described in the following three subsections. Since the 60-year projected operational cycles were used in determining that the design fatigue analyses remain valid for 60 years, the Metal Fatigue of Reactor Coolant Pressure Boundary Program must continue to be used to validate the assumptions used in the evaluations. Therefore, Class 1 components and piping fatigue TLAAs, except in certain specific cases described in the following three subsections, have been dispositioned in accordance with 10 CFR 54.21(c)(1)(i) and 10 CFR 54.21(c)(1)(iii).

A.3.3.1.1 Unit 2 RHR Piping and Unit 2 Charging Line

The RHR piping and the charging line cycles of operation are projected to exceed their respective design cycles during the period of extended operation. The Metal Fatigue of Reactor Coolant Pressure Boundary Program will be used to monitor the transient cycles for the RHR piping and the Unit 2 charging line. As required by the program, corrective actions will be taken (including reanalysis, repair or replacement) such that the design basis of the these components are not exceeded for the period of extended operation. Therefore, the RHR piping and the charging line fatigue TLAAs have been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

A.3.3.1.2 Unit 2 Steam Generator Manway Bolts and Tubes

BVPS was not able to demonstrate that the original design fatigue calculations remained valid through the period of extended operation for the following sub-components of the Unit 2 steam generators:

- Steam generator secondary manway bolts; and,
- Steam generator tubes (U-bend fatigue).

The Unit 2 steam generator secondary manway bolts and the steam generator tubes fatigue analyses are based on a 40-year life (i.e., to 2027). In the Extended Power Uprate T_{AVG} coastdown analysis for the secondary manway bolts, BVPS assumed that the Unit 2 steam generators will be replaced by the year 2027. In the Uprate analysis for the U-bends, BVPS assumed that the Unit 2 steam generators will be replaced by the year 2027. As part of the Steam Generator Tube Integrity Program, BVPS will perform a reanalysis, repair, or replacement of the affected components such that the design bases of the these components are not exceeded for the period of extended operation. Therefore, the steam generator secondary manway bolts and the steam generator tubes fatigue TLAAs have been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

A.3.3.1.3 Unit 2 Pressurizer

In 2000, the analysis of the Unit 2 pressurizer, lower shell and related components was revised to address revision to the insurge/outsurge transients identified by the Westinghouse Owners Group. Plant operating procedures were revised to follow the guidance of the Westinghouse Owners Group and to minimize the impact of potential insurges. In 2002, BVPS decided to further revise the operating procedures to optimize the plant shutdown and startup processes for Unit 2. The BVPS optimized procedures have been shown to meet all recommendations of the Westinghouse Owners Group and have virtually eliminated the potential for insurges. Next, the Extended Power Uprate Project evaluated the revised Uprate transients against the previous analysis. Since some operating parameters changed, BVPS revised the analysis of the Unit 2 pressurizer, lower shell and related components. In addition, the pressurizer spray nozzle, the safety valve nozzles, the pressure operated relief valve nozzle and the surge line nozzle were potentially impacted by the Pressurizer Weld Overlay Project. Weld overlay was performed during the Unit 2 Cycle 12 Refueling Outage (October - November 2006). Weld overlay for the surge nozzle is discussed in a supplement to the subject analysis. The cumulative usage factors associated with the Unit 2 pressurizer are less than 1.0. Since the 60-year projected operational cycles were used in determining that the pressurizer design fatigue analysis remains valid for 60 years, the Metal Fatigue of Reactor Coolant Pressure Boundary Program (Section B.2.27) must continue to be used to validate the assumptions used in the evaluation. In addition, the pressurizer insurge cycle assumptions used in the pressurizer analysis require validation for the period of extended operation. The Metal Fatigue of Reactor Coolant Pressure Boundary Program identifies the pressurizer insurge transient as a supplemental transient that requires monitoring. Therefore, the pressurizer fatigue TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

A.3.3.2 Non-Class 1 Fatigue Evaluations

A.3.3.2.1 Piping and In-Line Components

The design code for non-Class 1 piping and in-line components (e.g., fittings and valves) within the scope of license renewal is ANSI B31.1 or ASME III, Subsections NC and ND. These codes specify evaluation of cyclic secondary stresses (i.e., stresses due to thermal expansion and anchor movements) by applying stress range reduction factors against the allowable stress range (SA).

For those non-Class 1 components identified as subject to cracking due to fatigue, a review of system operating characteristics was conducted by BVPS to determine the approximate frequency of any significant thermal cycling. If the number of equivalent full-temperature cycles is below the limit used for the current design, the component is suitable for extended operation. If the number of equivalent full-temperature cycles exceeds the limit, evaluation of the individual stress calculations will be required.

BVPS evaluated the validity of this assumption for 60 years of plant operation. With the exception of the Unit 2 Emergency Diesel Generator (EDG) Air Start System, the results of this evaluation indicated that the thermal cycle assumption is valid and bounding for 60 years of operation. Therefore, the non-Class 1 piping fatigue TLAAs, with the exception of the EDG Air Start System fatigue TLAA, remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

The EDG Air Start System contains components potentially subject to fatigue. As part of the Metal Fatigue of Reactor Coolant Pressure Boundary Program, BVPS will perform an assessment to determine whether the full-temperature cycles limit would be exceeded for 60 years of operation. Corrective actions will be taken as appropriate (including reanalysis, repair or replacement), such that the full-temperature cycles limit of the EDG Air Start System is not exceeded for the period of extended operation. Therefore, the EDG Air Start System fatigue TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

A.3.3.2.2 Pressure Vessels, Heat Exchangers, Storage Tanks, Pumps, and Turbine Casings

Non-Class 1 pressure vessels, heat exchangers, storage tanks, pumps, and turbine casings are typically designed in accordance with ASME Section VIII or ASME Section III, Subsection NC or ND (e.g., Class 2 or 3). Some tanks and pumps are designed to other industry codes and standards (such as American Water Works Association and Manufacturer's Standardization Society), reactor designer specifications, and architect engineer specifications. Only ASME Section VIII, Division 2, and ASME Section III, Subsection NC-3200, design codes include fatigue design requirements. Due to the conservatism in ASME Section VIII Division 1 and ASME Section III NC-3100/ND-3000 detailed fatigue analyses are not required. If cyclic loading and fatigue usage could be significant, the component designer is expected to specify ASME Section VIII Division 2 or NC-3200. For components where there is no required fatigue analysis, cracking due to fatigue is not an aging effect requiring management.

Fatigue analysis is not required for ASME Section VIII Division I, Section III NC-3100 or ND vessels. It is also not required for NC/ND pumps and storage tanks (<15 psig). The design specification identifies the applicable design code for each component.

Only the Unit 2 non-regenerative (letdown), regenerative, and RHR heat exchangers were identified as having fatigue TLAAs, and are dispositioned as described in the following text.

A.3.3.2.2.1 Non-regenerative (Letdown) Heat Exchanger

The Unit 2 non-regenerative (letdown) heat exchanger is designed to ASME, Section III, Class C (tubes) and ASME, Section VIII, Division 1 (shell). The transients for the non-regenerative (letdown) heat exchanger are defined in Westinghouse Equipment Specification G-679150 [Reference A.3-9]. The fatigue analysis associated with the Unit 2 non-regenerative (letdown) heat exchanger is not bounding for 60 years of operation. The Metal Fatigue of Reactor Coolant Pressure Boundary Program will be used to monitor the Unit 2 non-regenerative (letdown) heat exchanger transients. As required by the program, corrective actions will be taken as appropriate (including reanalysis, repair or replacement), such that the design basis of the Unit 2 non-regenerative (letdown) heat exchanger is not exceeded for the period of extended operation. Therefore, the Unit 2 non-regenerative (letdown) heat exchanger fatigue TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

A.3.3.2.2.2 Regenerative Heat Exchanger

The Unit 2 regenerative heat exchanger was built to ASME, Section III, Class 2. The transients for the Unit 2 regenerative heat exchanger are defined in Westinghouse Equipment Specification G-679150. The fatigue analysis associated with the Unit 2 regenerative heat exchanger is not bounding for 60 years of operation. The Metal Fatigue of Reactor Coolant Pressure Boundary Program will be used to monitor the regenerative heat exchanger transients. As required by the program, corrective actions will be taken as appropriate (including reanalysis, repair or replacement), such that the design basis of the regenerative heat exchanger is not exceeded for the period of extended operation. Therefore, the Unit 2 regenerative heat exchanger fatigue TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

A.3.3.2.2.3 Residual Heat Removal (RHR) Heat Exchangers

The tube side of the Unit 2 RHR heat exchangers were designed in accordance with ASME Section III, Class 2. The shell side of these heat exchangers were designed in accordance with ASME Section III, Class 3. The transients for the RHR heat exchangers are defined in Westinghouse Equipment Specification G-679150. The fatigue analyses associated with the RHR heat exchangers are not bounding for 60 years of operation. The Metal Fatigue of Reactor Coolant Pressure Boundary Program will be used to monitor the Unit 2 RHR heat exchangers transients. As required by the program, corrective actions will be taken as appropriate (including reanalysis, repair or replacement), such that the design basis of the Unit 2 RHR heat exchangers are not exceeded for the period of extended operation. Therefore, the Unit 2 RHR heat exchangers fatigue TLAAs have been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

A.3.3.3 Generic Industry Issues on Fatigue

This section addresses the BVPS fatigue TLAAs associated with NRC Bulletins 88-08 and 88-11. In addition, this section addresses the effects of the primary coolant environment on fatigue life.

A.3.3.3.1 Thermal Stresses in Piping Connected to Reactor Coolant System (NRC Bulletin 88-08)

NRC Bulletin 88-08, *Thermal Stresses in Piping Connected to Reactor Coolant Systems* [Reference A.3-10], requested that licensees: (1) review their RCS to identify any connected unisolable piping that could be subjected to temperature distributions which would result in unacceptable thermal stresses and any unisolable sections of piping connected to the RCS that may have been subjected to excessive thermal stresses, and, (2) take action, where such piping is identified, to ensure that the piping will not be subjected to unacceptable thermal stresses. There are no specific TLAA associated with the Unit 2 responses to NRC Bulletin 88-08, with the exception of the ASME Class 2 RHR line analysis.

The Unit 2 RHR line stratification analysis required a detailed fatigue evaluation to demonstrate compliance with the design code of record (ASME Section III). Based on temperature data established in response to NRC Bulletin 88-08, a conservative thermal stratification load case was developed. Typical cycle periods for the thermal stratification events on the Unit 2 RHR lines were 6 to 8 days, which equated to approximately 2000 cycles for a 40-year plant life (assuming the stratification occurred continuously). A bounding thermal stratification load assuming 7000 cycles was incorporated into the fatigue analysis as an additional load. Projecting the identified stratification cycles for a 60-year plant life results in 3000 cycles. The 7000 cycles used in the fatigue analysis bounds the 60-year projected cycles. Therefore, the Unit 2 RHR line fatigue TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

A.3.3.3.2 Pressurizer Surge Line Thermal Stratification (NRC Bulletin 88-11)

NRC Bulletin 88-11, *Pressurizer Surge Line Thermal Stratification* [Reference A.3-11], required a plant-specific or generic analysis demonstrating that the pressurizer surge line meets the applicable design code requirements considering the effects of thermal stratification.

Pressurizer surge line stratification first became apparent at Unit 2 during hot functional testing, and was a predecessor to NRC Bulletin 88-11. Additional instrumentation was temporarily installed to monitor pipe and fluid conditions. From this data, BVPS revised the surge line ASME Section III analysis of record to evaluate stress and fatigue effects.

Subsequently, BVPS contracted Westinghouse to perform a complete reanalysis of the surge line, accounting for thermal stratification and striping. WCAP-12093, *Evaluation of Thermal Stratification for the Beaver Valley Unit 2 Pressurizer Surge Line* [Reference A.3-12], was submitted to the NRC to address both leak-before-break (LBB) requirements and NRC Bulletin 88-11 concerns for the surge line. The NRC accepted [Reference A.3-13] WCAP-12093 as meeting the required actions of NRC Bulletin 88-11, and demonstrating that the effects of thermal stratification do not result in the pressurizer surge line exceeding design Code allowable limits.

WCAP-12093 determined the effect of thermal stratification through the imposition of defined thermal stratification cycles upon the stress and fatigue evaluations. The stratification cycles incorporated into the cumulative usage factor determination are defined by the 200 heatup and cooldown design transients.

Therefore, these NRC Bulletin 88-11 analyses are TLAAs in accordance with 10 CFR 54.3.

WCAP-12093 was reviewed for impact due to extended power uprate. A detailed analysis was performed at the controlling location (reactor coolant loop nozzle) to account for temperature effects due to the power uprate. A new cumulative usage factor was calculated and demonstrated to remain less than the Code allowable limit of 1.0.

The 200 heatup and cooldown transients were determined to remain bounding for the period of extended operation. Therefore, the Unit 2 pressurizer surge line fatigue TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

A.3.3.3.3 Effects of Primary Coolant Environment on Fatigue Life

Test data indicate that certain environmental conditions (such as temperature, oxygen content, and strain rate) in the primary systems of light water reactors could result in greater susceptibility to fatigue than would be predicted by fatigue analyses based on the ASME Section III design fatigue curves. One NRC study, documented in NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components* [Reference A.3-14], applied the fatigue design curves that incorporated environmental effects to several plant designs. The results of studies performed on this topic, including NUREG/CR-6260, were summarized in Generic Safety Issue (GSI)-190, *Fatigue Evaluation of Metal Components for 60-Year Plant Life* [Reference A.3-15]. In closing GSI-190, regarding the effects of a reactor water environment on fatigue life, the NRC concluded that licensees should address the effects of the coolant environment on component fatigue life as aging management programs are formulated in support of license renewal.

The Unit 2 reactor coolant pressure boundary piping is designed to ASME Section III, and is therefore classified as a newer-vintage Westinghouse plant.

Section 5.4 of NUREG/CR-6260 identified the following component locations as representative for environmental effects for newer-vintage Westinghouse plants. These locations and the subsequent calculations are directly relevant to Unit 2 and include the:

- Reactor vessel shell and lower head (shell-to-head transition);
- Reactor vessel inlet and outlet nozzles;
- Pressurizer surge line (hot leg nozzle safe end);
- RCS piping charging system nozzle (knuckle region);
- RCS piping safety injection nozzle (knuckle region); and,
- RHR system piping (inlet piping transition).

The NUREG/CR-6260 locations were evaluated using the guidance of NUREG/CR-6583, *Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low Alloy Steels* [Reference A.3-16], and NUREG/CR-5704, *Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels* [Reference A.3-17]. These reports describe the use of a fatigue life correction factor (F_{en}) to express the effects of the reactor coolant environment upon the material fatigue life. The expression for F_{en} was determined through experimental and statistical data. F_{en} for carbon and low alloy steel is a function of fluid service temperature, material sulfur content, fluid dissolved oxygen, and strain rate. For austenitic stainless steel, F_{en} is a function of fluid service temperature, fluid dissolved oxygen, and strain rate. The cumulative usage factor which includes environmental effects (U_{env}) is determined from the existing 60-year cumulative usage factor (U_{60}) through the use of the fatigue life correction factor:

$$U_{env} = U_{60} * F_{en}$$

To demonstrate acceptable fatigue life including environmental effects, the cumulative usage factor, which includes environmental effects, should remain less than design code allowables (i.e., $U_{env} \leq 1.0$). Therefore, F_{en} was applied to the cumulative usage factors at the Unit 2 NUREG/CR-6260 locations and compared to the design code allowable limit.

At three locations (pressurizer surge line, charging system nozzle, and RHR system piping), U_{env} exceeded the design code allowable limit of 1.0. For these locations, BVPS will implement one or more of the following as required by the Metal Fatigue of Reactor Coolant Pressure Boundary Program:

1. Further refinement of the fatigue analyses to lower the predicted CUFs to less than 1.0;
2. Management of fatigue at the affected locations by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method acceptable to the NRC); or,
3. Repair or replacement of the affected locations.

Should BVPS select the option to manage environmentally-assisted fatigue during the period of extended operation, details of the aging management program, such as scope, qualification, method, and frequency, will be submitted to the NRC prior to the period of extended operation. Therefore, the TLAAs associated with the pressurizer surge line, charging system nozzle, and RHR system piping have been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

The cumulative usage factors including environmental fatigue at the other locations (reactor vessel shell and lower head, reactor vessel inlet and outlet nozzles, and safety injection nozzle), have been demonstrated to remain less than the design code allowable limit of 1.0 for the period of extended operation. Therefore, the TLAAs associated with these locations have been dispositioned in accordance with 10 CFR 54.21(c)(1)(ii).

A.3.4 ENVIRONMENTAL QUALIFICATION (EQ) OF ELECTRIC EQUIPMENT

The BVPS existing Environmental Qualification (EQ) of Electric Components Program manages component thermal, radiation and cyclical aging, as applicable, through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods. As required by 10 CFR 50.49, environmental qualification components not qualified for the current license term are to be refurbished, replaced, or have their qualification extended prior to reaching the aging limits established in the evaluation. The Environmental Qualification of Electric Components Program ensures that these environmental qualification components are maintained in accordance with their qualification bases. Aging evaluations for environmental qualification components that specify a qualification of at least 40 years are time-limited aging analyses for license renewal.

The Environmental Qualification (EQ) of Electric Components Program is an existing program established to meet BVPS commitments for 10 CFR 50.49. Continued implementation of the Environmental Qualification (EQ) of Electrical Components Program provides reasonable assurance that the aging effects will be managed and that the in-scope EQ components will continue to perform their intended function(s) for the period of extended operation. The effects of aging will be managed by the program in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

A.3.5 Containment Liner Plate, Metal Containment, and Penetrations Fatigue

Several potential TLAA associated with the Containment structure were identified and are summarized in the following sections:

- Containment Liner Fatigue (Section A.3.5.1)
- Containment Liner Corrosion Allowance (Section A.3.5.2)
- Containment Liner Penetration Fatigue (Section A.3.5.3)

A.3.5.1 Containment Liner Fatigue

The Containment liner was designed using the 1971 Edition of ASME Section III as a design guideline using stress limits and fatigue criteria based on the rules for code classes MC and 1. As such, a detailed analysis for fatigue is not required if six specific requirements are met as defined in ASME Section III, NB-3222.4(d). This exemption analysis was performed for the 40-year anticipated stress cycles of differential pressure due to normal operation (100 cycles), differential temperature due to normal operation (400 cycles), and ½ safe shutdown earthquake (operational basis earthquake) (100 cycles). To address these 40-year cycles, a re-evaluation of the six fatigue exemption requirements utilizing anticipated 60-year stress cycles was performed. The anticipated occurrences of these cycles are described in Table 3.8-9 of the Unit 2 UFSAR as follows:

- 150 stress cycles of differential pressure loading assuming 2.5 refueling cycles per year on a 60-year span;
- 600 stress cycles of loading due to thermal expansion resulting from exposure to the differential temperature between operating and seasonal refueling temperatures based on 10 such variations per year on a 60-year span; and,
- 150 cycles of operational basis earthquake, which is an assumed number of cycles of this type of earthquake for a 60-year span.

The result of this evaluation determined that the specified normal conditions through the period of extended operation continue to satisfy the requirement for exemption from analysis for cyclic operation. Therefore, the Unit 2 Containment liner fatigue TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(ii).

A.3.5.2 Containment Liner Corrosion Allowance

The Reactor Containment Building has a continuously welded carbon steel liner which acts as a leak-tight membrane. The cylindrical portion of the liner is 3/8 inch thick, the hemispherical dome is ½ inch thick, and the flat floor liner covering the concrete mat is ¼ inch thick. The floor liner plate is covered with approximately two feet of reinforced concrete. All welded seams were originally covered with continuously welded leak test channels that were installed to facilitate leak testing of welds during liner erection. Since initial construction, several test channels have been removed. Channels in the hemispherical dome and Containment mat are covered with concrete while those on the cylindrical liner wall are exposed. Test ports that were provided for leak testing were sealed with vent plugs after the completion of the testing. These plugs were to remain in place during subsequent Type A leak rate testing.

During the second refueling outage for Unit 2 in 1990, the results of an inspection performed prior to the Type A Containment leakage rate test showed that 25 test channel vent plugs were missing. The missing test channel vent plugs allowed moisture and condensation inside the test channels, leading to minor corrosion of the liner. BVPS evaluated the test channels to determine the impact to the Containment liner, and submitted the results of the evaluations to the NRC as Amendments 165 and 47, Unit 1 and Unit 2 respectively, to the operating licenses. These amendments were approved by the NRC and documented in an SER [Reference A.3-18]. After further evaluation, it was concluded that these initial evaluations contained some nonconservative assumptions with regard to the corrosion rates in the test channels. BVPS took corrective action to arrest the corrosion rate in the affected test channels, including inerting and sealing the test channels. The further evaluation and corrective actions are documented in a 1992 Letter to the NRC [Reference A.3-19]. These corrosion rate analyses meet the 10 CFR 54.3 requirements as TLAAs and must be evaluated for the period of extended operation.

The minimum required thickness for the Containment liner has been determined for the various portions of the liner. The limiting liner portion is the liner floor plate, which has a fabrication thickness of 0.25 inches and a minimum required thickness of 0.125 inches. Thus, the corrosion allowance is 0.125 inches (125 mils). The inerting and sealing of the test channels significantly reduced the theoretical corrosion rates in the channels. The total estimated penetration due to corrosion of the inerted channel was estimated at 82.7 mils for 43 years of plant operation. The maximum expected corrosion rate for the carbon steel liner in this low oxygen environment was determined to be 0.39 mils per year. Therefore, projecting the expected corrosion penetration with the maximum expected corrosion rate to the end of the period of extended operation results in an additional 7.8 mils of corrosion. Adding this to the previous expected corrosion penetration depths yields 90.5 mils of corrosion penetration. This result is well within the corrosion allowance of 125 mils.

Therefore, the Unit 2 Containment liner corrosion analysis has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A.3.5.3 Containment Liner Penetration Fatigue

A.3.5.3.1 Containment Process Piping Penetrations

Unit 2 process piping penetrations are designed and analyzed to the 1971 Edition through 1972 Winter Addenda of ASME Section III, Division 1, Class 2 (i.e., Subsection NC), which complies with the process piping system requirements of which these penetrations are a part. The penetrations are further analyzed to the more stringent Class MC (i.e., Subsection NE) requirements. Section III, Division 1, Class 2 requirements include a stress range reduction factor which accounts for an assumed number of thermal cycles. Additionally, Section III, Division 1, Class MC states that any portions not satisfying the fatigue exemption as described in Subsection NB-3222(d) require further fatigue evaluation. These thermal cycles and fatigue exemptions are based on a design number of cycles for the plant life. As such, the Unit 2 piping penetration analyses are classified as TLAAs and require disposition for the period of extended operation.

For the Unit 2 process piping penetrations identified as subject to cracking due to fatigue, a review of system operating characteristics was conducted to determine the approximate frequency of significant thermal cycling. If the number of equivalent full-temperature cycles is below the limit used for the current design, the component is suitable for extended operation. If the number of equivalent full-temperature cycles exceeds the limit, evaluation of the individual stress calculations will be required.

BVPS evaluated the validity of this assumption for 60 years of plant operation. The results of this evaluation indicate that the thermal cycle assumption is valid and bounding for 60 years of operation. Therefore, these piping penetration fatigue analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.3.5.3.2 Equipment Hatch

The equipment hatch and integral emergency airlock are designed and analyzed in accordance with ASME Section III, Division 1, Subsection NE (Class MC). Subsection NE states that any portions not satisfying the fatigue exemption as described in Subsection NB-3222(d) require further fatigue evaluation. Therefore, a fatigue exemption was completed for the Unit 2 equipment hatch in accordance with Subsection NB-3222(d). This exemption was based on assumed cycles for a 40 year life, namely 10 pressurization events due to LOCA, and 80 cycles of startup and shutdown. It is highly unlikely that Unit 2 will reach 10 pressurization events due to LOCA for 60 years of operation. The assumption of 80 cycles of startup and shutdown is not bounding for 60 years of operation. A reanalysis was performed using 240 startup and shutdown cycles that bounds the number of projected cycles for the period of extended operation. Therefore, the equipment hatch fatigue TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(ii).

A.3.5.3.3 Fuel Transfer Tube

The fuel transfer tube pipe was analyzed to ASME Section III, Class 2. The analysis for the fuel transfer tube pipe uses a stress range reduction factor of 1.0 (<7,000 cycles). However, as the fuel transfer tube pipe experiences operational cycles only during refueling, the fuel transfer tube pipe experiences essentially no thermal cycles. The existing fuel transfer tube pipe fatigue TLAA remains valid through the period of extended operation. Therefore, the fuel transfer tube pipe fatigue TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

The fuel transfer tube bellows were analyzed to ASME Section III, Class MC. The bellows stress analyses determined acceptability based on the bellows experiencing displacements due to a design basis earthquake. The assumed design cycles were 600. This number of design basis earthquake cycles is highly unlikely to occur during the period of extended operation. The fuel transfer tube bellows fatigue TLAAs remain valid through the period of extended operation. Therefore, the fuel transfer tube bellows fatigue TLAAs have been dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

A.3.6 OTHER PLANT-SPECIFIC TIME-LIMITED AGING ANALYSES

The plant-specific TLAAs summarized in this section include:

- Leak Before Break (Section A.3.6.1)
- High Energy Line Break Postulation (Section A.3.6.2)
- Settlement Of Structures (Section A.3.6.3)
- Crane Load Cycles (Section A.3.6.4)

A.3.6.1 Leak Before Break

Leak before break (LBB) analyses evaluate postulated flaw growth in piping to alter the structural design basis. BVPS has determined that the fatigue crack growth analysis is a TLAA that requires disposition for license renewal.

For the LBB analyses discussed in the following three subsections, the only consideration that could be influenced by time is the accumulation of actual fatigue transient cycles. The cycle assumptions used in the analyses are conservative compared to the BVPS original design transients. The BVPS original design basis transients including design cycles are identified in Table 3.9N-1 of the UFSAR. BVPS has reviewed the design cycles against the 60-year projected operational cycles and has determined that the design cycles are bounding for the period of extended operation. Since the 60-year projected operational cycles were used in determining that the fatigue crack growth analyses remain valid for 60 years, the Metal Fatigue of Reactor Coolant Pressure Boundary Program must continue to be used to validate the assumptions used in the evaluations. Therefore, the LBB TLAAs have been dispositioned in accordance with 10 CFR 54.21(c)(1)(i) and 10 CFR 54.21(c)(1)(iii).

A.3.6.1.1 Main Coolant Loop Piping Leak Before Break

The current LBB evaluation for the main coolant loop piping is documented in WCAP-11923, *Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Beaver Valley Unit 2 After Reduction of Snubbers* [Reference A.3-20]. This evaluation was approved by the NRC in an SER [Reference A.3-21] in 1991.

A.3.6.1.2 Pressurizer Surge Line Piping Leak Before Break

The current LBB evaluation for the pressurizer surge line piping is documented in WCAP-12093, *Evaluation of Thermal Stratification for the Beaver Valley Unit 2 Pressurizer Surge Line* [Reference A.3-22]. This evaluation (including Supplements 1 and 2) was approved by the NRC in an SER [Reference A.3-23] in 1990. These analyses were based on a maximum temperature difference of 315°F between the pressurizer and the hot leg. Subsequent to the 1990 SER, a system temperature difference of approximately 360°F was experienced in the plant during heatup. To address this issue, WCAP-12093-P, Supplement 3, *Evaluation of Pressurizer Surge Line Transients Exceeding 320°F for Beaver Valley Unit 2* [Reference A.3-24], was prepared and submitted to the NRC. This evaluation was approved by the NRC in an SER [Reference A.3-25] in 1991.

A.3.6.1.3 Branch Line Piping Leak Before Break

The Unit 2 branch line piping LBB analyses were approved by the NRC in NUREG-1057, Supplement No. 4, *Safety Evaluation Report Related to the Operation of Beaver Valley Power Station Unit 2* [Reference A.3-26].

A.3.6.2 High Energy Line Break Postulation

In accordance with 10 CFR 50, General Design Criterion No. 4, *Environmental and Missile Design Bases*, special measures have been taken in the design and construction of Unit 2 to protect SSCs required to place the reactor in a safe cold shutdown condition from the dynamic effects associated with the postulated rupture of piping.

For the Class 1 systems, Regulatory Guide 1.46, *Protection Against Pipe Whip Inside Containment* [Reference A.3-27], states that postulated break locations be determined, in part, using any intermediate locations between terminal ends where the cumulative usage factor derived from the piping fatigue analysis under the loadings associated with specified seismic events and operational plant conditions exceeded 0.1. These fatigue evaluations are TLAAs since they are based on a set of fatigue transients that are based on the life of the plant.

The cycle assumptions used in the fatigue analyses are conservative compared to the BVPS original design transients [Reference A.3-28]. The BVPS original design basis transients including design cycles are identified in Table 3.9N-1 of the UFSAR. BVPS has reviewed the design cycles against the 60-year projected cycles and has determined that the design cycles are bounding for the period of extended operation. Since the 60-year cycle projections were used in determining that the fatigue analyses remains valid for 60 years, the Metal Fatigue of Reactor Coolant Pressure Boundary Program must continue to be used to validate the assumptions used in the evaluations. Therefore, the piping fatigue analyses used for determining the postulation of break locations in Class 1 lines remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i) and 10 CFR 54.21(c)(1)(iii).

A.3.6.3 Settlement Of Structures

The foundation soils in the main plant area consist of compacted select granular fill and medium dense to dense in situ granular soils. Site subsurface profiles within the plant area are discussed in the UFSAR. Total static settlement of the plant structures founded on granular soils was assumed to consist of two components: an elastic component, and a time-dependent component, which was assumed to be equal in magnitude to the elastic component. Each in-scope plant structure typically has a shake space between it and any adjacent structures to allow independent movement in the event of earthquake loading. These shake spaces also allow for differential settlement between plant structures. Such settlement can affect safety-related piping that penetrates the structure.

Observed settlement data was used to predict settlement of structures that are penetrated by piping. The settlement predictions were based on an assumed 40-year plant life. Stress analyses for affected piping include stresses that would be imposed by the predicted settlement. Therefore, the predicted settlement values of plant structures are used in the design stress analyses of various piping systems which span structures or exit structures into the surrounding soil (buried piping). The settlement assumptions are based on projected 40 year settlement values and, as such, the piping stress analyses that use these settlement assumptions are TLAAs which must be dispositioned for the period of extended operation.

As documented in UFSAR Section 2.5.4.13, the settlement of each Category I structure was monitored during construction, and will be monitored throughout the life of the plant until the settlement of a particular structure has been determined to be stable as defined by the Settlement Monitoring Program (Section A.1.37). For such structures, settlement monitoring is then discontinued. The Settlement Monitoring Program provides the requirements to measure the settlement of structures at selected locations. If the settlement of a structure exceeds that anticipated, a review of current analysis (as it relates to the integrity of the structure and the maintenance of settlement assumptions in the associated piping stress analyses) is required.

The Settlement Monitoring Program ensures that the current 40-year settlement assumptions in the pipe fatigue analyses are maintained for the period of extended operation. Therefore, the piping fatigue TLAAs have been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

A.3.6.4 Crane Load Cycles

In the response to NUREG-0612, *Control of Heavy Loads at Nuclear Power Plants* [Reference A.3-29], BVPS determined that three cranes, the polar crane (2CRN-201), spent fuel cask trolley (2MHF-CRN215), and the moveable platform and hoists (2MHF-CRN227), were designed to comply with Crane Manufacturers Association of America Specification #70 (CMAA-70), *Specifications for Electric Overhead Traveling Cranes* [Reference A.3-30]. Therefore, these cranes have a TLAA associated with their design calculations.

These cranes may conservatively be classified as Service Class A cranes. The total load cycles and mean effective load factors for the cranes have been estimated for the period of extended operation. Even using conservative estimates, total load cycles are well below 20,000, and mean effective load factors are maintained within or below the Service Class A bounds (0.35 - 0.53) for 60 years. Therefore, crane allowable stress ranges as defined in CMAA-70 will remain valid through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A.3.7 APPENDIX A.3 REFERENCES

A.3-1 [BVPS License Renewal Application – later].

A.3-2 [NRC SER for BVPS License Renewal – later].

A.3-3 10 CFR 54, *Requirements for Renewal of Operating Licenses for Nuclear Power Plants*.

A.3-4 WCAP-16527-NP, *Analysis of Capsule X from First Energy Nuclear Operating Company Beaver Valley Unit 2 Reactor Vessel Radiation Surveillance Program*, Rev. 0.

A.3-5 WCAP-16527-NP Supplement 1, *Analysis of Capsule X from First Energy Company Beaver Valley Unit 2 Reactor Vessel Radiation Surveillance Program*, June 2007.

A.3-6 Regulatory Guide 1.190, *Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence*, March 2001.

A.3-7 10 CFR 50, *Domestic Licensing of Production and Utilization Facilities*.

A.3-8 Regulatory Guide 1.99, *Radiation Embrittlement of Reactor Vessel Materials*, Rev. 2.

A.3-9 Westinghouse Equipment Specification G-679150, *Auxiliary Heat Exchangers*, Rev. 1.

A.3-10 NRC Bulletin 88-08, *Thermal Stresses in Piping Connected to Reactor Coolant Systems*, June 22, 1988, (including Supplements 1 and 2).

A.3-11 NRC Bulletin 88-11, *Pressurizer Surge Line Thermal Stratification*, December 20, 1988.

A.3-12 WCAP-12093, Evaluation of Thermal Stratification for the Beaver Valley Unit 2 Pressurizer Surge Line, Rev. 0, including Supplements 1, 2, and 3.

A.3-13 Tam, Peter S. (NRC), Letter to J. D. Sieber (BVPS), *Beaver Valley Unit 2 - Completion of Review on Pressurizer Surge Line Thermal Stratification (TAC No. 72111)*, January 18,1990.

A.3-14 NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components*, February 1995.

A.3-15 Generic Safety Issue (GSI)-190, *Fatigue Evaluation of Metal Components for 60-Year Plant Life*, Rev. 2.

A.3-16 NUREG/CR-6583, *Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low Alloy Steels*, February 1998.

A.3-17 NUREG/CR-5704, *Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels*, March 1999.

A.3-18 De Agazio, Albert W. (NRC), Letter to J. D. Sieber (BVPS), *Beaver Valley Units 1 And 2 - Issuance of Amendments 165 and 47: Containment Structural Integrity - Change Request Nos. 181/45*, June 23, 1992.

A.3-19 Sieber, J. D. (BVPS), Letter to NRC, *Beaver Valley Power Station, Unit No. 1 and No. 2, BV-1 Docket No. 50-334, License No. DPR-66, BV-2 Docket No. 50-412, License No. NPF-73, Revision to SER for Amendments 165 and 47*, December 30, 1992.

A.3-20 WCAP-11923, *Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Beaver Valley Unit 2 After Reduction of Snubbers*, September 1988.

A.3-21 De Agazio, Albert W. (NRC), Letter to J. D. Sieber (BVPS), *Elimination of Dynamic Effects of Postulated Pressurizer Surge Line Rupture and Elimination of Reactor Coolant System Component Support Snubbers*, April 8, 1991.

A.3-22 WCAP-12093, *Evaluation of Thermal Stratification for the Beaver Valley Unit 2 Pressurizer Surge Line*, December 1988.

A.3-23 Tam, Peter S. (NRC), Letter to J. D. Sieber (BVPS), *Beaver Valley Unit 2 - Completion of Review on Pressurizer Surge Line Thermal Stratification*, January 18, 1990.

A.3-24 Sieber, J. D. (BVPS), Letter to NRC, *Beaver Valley Power Station, Unit No. 2, Docket No. 50-412, License No. NPF-73, Primary Component Support Snubber Elimination*, August 10, 1990.

A.3-25 De Agazio, Albert W. (NRC), Letter to J. D. Sieber (BVPS), *Elimination of Dynamic Effects of Postulated Pressurizer Surge Line Rupture and Elimination of Reactor Coolant System Component Support Snubbers*, April 8, 1991.

A.3-26 NUREG-1057, Supplement No. 4, *Safety Evaluation Report Related to the Operation of Beaver Valley Power Station Unit 2; Docket No. 50-412 Duquesne Light Company*, March 1987.

A.3-27 Regulatory Guide 1.46, *Protection Against Pipe Whip Inside Containment*, May 1973.

A.3-28 NRC Letter, Timothy G. Colburn (NRC), to James H. Lash (FENOC), *Beaver Valley Power Station, Unit 1 and Unit 2 (BVPS-1 and 2) - Issuance of Amendment Regarding the 8-Percent Extended Power Uprate*, July 19, 2006.

A.3-29 NUREG-0612, *Control of Heavy Loads at Nuclear Power Plants*, July 1980.

A.3-30 Crane Manufacturers Association of America Specification #70 (CMAA-70), *Specifications for Electric Overhead Traveling Cranes*, Revised 1983.

A.4 Unit 1 License Renewal Commitments

Table A.4-1 identifies those actions committed to by FENOC for BVPS Unit 1 in the BVPS License Renewal Application (LRA). These regulatory commitments will be tracked within the FENOC regulatory commitment management program. Any other actions discussed in the LRA represent intended or planned actions by FENOC. These other actions are described only as information and are not regulatory commitments. This list will be revised as necessary in subsequent amendments to reflect changes resulting from NRC audit questions and BVPS responses to NRC requests for additional information.

Table A.4-1
Unit 1 License Renewal Commitments

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
1	Implement the Buried Piping and Tanks Inspection Program as described in LRA Section B.2.8.	Will be implemented within the 10 years prior to January 29, 2016	LRA	A.1.8 B.2.8
2	Enhance the Closed-Cycle Cooling Water System Program to: • Add the diesel-driven fire pump (Unit 1 only) to the program; • Detail performance testing of heat exchangers and pumps, and provide direction to perform visual inspections of system components; • Identify closed-cycle cooling water system parameters that will be trended to determine if heat exchanger tube fouling or corrosion product buildup exists; • Control performance tests and perform visual inspections at the required frequency.	January 29, 2016	LRA	A.1.9 B.2.9

Table A.4-1
Unit 1 License Renewal Commitments
(continued)

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
3	Implement the Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements One-Time Inspection Program as described in LRA Section B.2.10.	Will be implemented within the 10 years prior to January 29, 2016	LRA	A.1.10 B.2.10
4	Implement the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program as described in LRA Section B.2.11.	January 29, 2016	LRA	A.1.11 B.2.11
5	Implement the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program as described in LRA Section B.2.12.	January 29, 2016	LRA	A.1.12 B.2.12
6	Implement the External Surfaces Monitoring Program as described in LRA Section B.2.15.	January 29, 2016	LRA	A.1.15 B.2.15
7	Enhance the Fire Protection Program to: • Include a new attachment in the BVPS Fire Protection Program administrative procedure to address the Fire Protection Systems that are in scope for license renewal purposes; • Provide details of the NUREG-1801 inspection and testing guidelines, the plant implementation strategy, surveillance test and inspection frequencies, and affected implementing procedure(s); and,	January 29, 2016	LRA	A.1.16 B.2.16

**Table A.4-1
Unit 1 License Renewal Commitments
(continued)**

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
7, cont.	• Provide inspection guidance details to include degradation such as concrete cracking and spalling, and loss of material of fire barrier walls, ceilings and floors that may affect the fire rating of the assembly or barrier.			
8	Enhance the Fire Water System Program to: • Include a program requirement to perform flow test or inspection of all accessible fire water headers and piping during the period of extended operation at an interval determined by the Fire Protection System Engineer; • Include a program requirement that a representative number of fire water piping locations be identified if piping visual inspections are used as an alternative to non-intrusive testing; • Include a program requirement which allows test or inspection results from an accessible section of pipe to be extrapolated to an inaccessible, but similar section of pipe. If no similar section of accessible pipe is available, then alternative testing or inspection activities must be used; • Include a program requirement that, at least once prior to the period of extended operation, all accessible Fire Protection headers and piping shall be flow tested in accordance with NFPA 25 or visually/ultrasonically inspected;	Will be implemented within the 10 years prior to January 29, 2016	LRA	A.1.17 B.2.17

Table A.4-1
Unit 1 License Renewal Commitments
(continued)

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
8, cont.	• Include steps in the program procedure that require testing or replacement of sprinkler heads that will have been in service for 50 years; and, • Include a program requirement to perform a fire water subsystem internal inspection any time a subsystem (including fire pumps) is breached for repair or maintenance.			
9	Enhance the Flux Thimble Tube Inspection Program to: • Include a requirement in the program procedure to state that, if a flux thimble tube cannot be inspected over the tube length (tube length that is subject to wear due to restriction or other defect), and cannot be shown by analysis to be satisfactory for continued service, the thimble tube must be removed from service to ensure the integrity of the Reactor Coolant System pressure boundary.	January 29, 2016	LRA	A.1.19 B.2.19
10	Enhance the Fuel Oil Chemistry Program to: • Revise the implementing procedure for sampling and testing the diesel-driven fire pump fuel oil storage tank (Unit 1 only) to include a test for particulate and accumulated water in addition to the test for sediment and water; and,	January 29, 2016	LRA	A.1.20 B.2.20

Table A.4-1
Unit 1 License Renewal Commitments
(continued)

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
10, cont.	• Generate a new implementing procedure for sampling and testing the security diesel generator fuel oil day tank (Common) for accumulated water, particulate contamination, and sediment / water.			
11	Implement the Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program as described in LRA Section B.2.21.	January 29, 2016	LRA	A.1.21 B.2.21
12	Implement the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program as described in LRA Section B.2.22.	January 29, 2016	LRA	A.1.22 B.2.22
13	Enhance the Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems Program to: • Include guidance in the program administrative procedure to inspect for loss of material due to corrosion on Unit 1 crane and trolley structural components and rails; and, • Include guidance in the crane and hoist inspection procedures to inspect for loss of material due to corrosion on Unit 1 crane and trolley structural components and rails or extendable arms, as appropriate.	January 29, 2016	LRA	A.1.23 B.2.23

Table A.4-1
Unit 1 License Renewal Commitments
(continued)

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
14	Enhance the Masonry Wall Program to: • Include in program scope additional masonry walls identified as having aging effects requiring management for license renewal.	January 29, 2016	LRA	A.1.25 B.2.25
15	For the Nickel-Alloy Nozzles and Penetrations Program, regarding activities for managing the aging of nickel-alloy components and nickel-alloy clad components susceptible to primary water stress corrosion cracking - PWSCC (other than upper reactor vessel closure head nozzles and penetrations), BVPS commits to develop a plant-specific aging management program that will implement applicable: 1. NRC Orders, Bulletins and Generic Letters; and, 2. Staff-accepted industry guidelines.	January 29, 2016	LRA	A.1.28 B.2.28
16	Implement the One-Time Inspection Program as described in LRA Section B.2.30.	Will be implemented within the 10 years prior to January 29, 2016	LRA	A.1.30 B.2.30
17	Implement the One-Time Inspection of ASME Code Class 1 Small-Bore Piping Program as described in LRA Section B.2.31.	Will be implemented within the 10 years prior to January 29, 2016	LRA	A.1.31 B.2.31

Table A.4-1
Unit 1 License Renewal Commitments
(continued)

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
18	For the PWR Vessel Internals Program, regarding activities for managing the aging of Reactor Vessel internal components and structures, BVPS commits to: 1. Participate in the industry programs applicable to BVPS Unit 1 for investigating and managing aging effects on reactor internals; 2. Evaluate and implement the results of the industry programs as applicable to the BVPS Unit 1 reactor internals; and, 3. Upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for the BVPS Unit 1 reactor internals to the NRC for review and approval.	January 29, 2014	LRA	A.1.33 B.2.33
19	Implement the Selective Leaching of Materials Program as described in LRA Section B.2.36.	January 29, 2016	LRA	A.1.36 B.2.36
20	Enhance the Structures Monitoring Program to: • Include in program scope additional structures and structural components identified as having aging effects requiring management for license renewal; • Include inspection guidance in program implementing procedures to detect significant cracking in concrete surrounding the anchors of vibrating equipment;	January 29, 2016	LRA	A.1.39 B.2.39

Table A.4-1
Unit 1 License Renewal Commitments
(continued)

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
20, cont.	• Include a requirement in program procedures to perform opportunistic inspections of normally inaccessible below-grade concrete when excavation work uncovers a significant depth; • Include a requirement in program procedures to perform periodic sampling of groundwater for pH, chloride concentration, and sulfate concentration; and, • Include a requirement in program procedures to monitor elastomeric materials used in seals and sealants, including compressible joints and seals, waterproofing membranes, etc., associated with in-scope structures and structural components for cracking and change in material properties.			
21	Implement the Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program as described in LRA Section B.2.40.	January 29, 2016	LRA	A.1.40 B.2.40
22	Implement the Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program as described in LRA Section B.2.41.	January 29, 2016	LRA	A.1.41 B.2.41
23	Enhance the Water Chemistry Program to: • Change BVPS frequency for reactor coolant silica monitoring to once per week for Operational Modes 1 and 2, and once per day during heatup in Operational Modes 3 and 4 to be consistent with EPRI guidelines.	January 29, 2016	LRA	A.1.42 B.2.42

Table A.4-1
Unit 1 License Renewal Commitments
(continued)

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
24	Documentation of a flux reduction program for Unit 1 will be submitted in accordance with the requirements of 10 CFR 50.61.	In accordance with the requirements of 10 CFR 50.61	LRA	A.2.2.2 4.2.2
25	Of the NUREG/CR-6260 locations, the U_{env} (60 years) of the Unit 1 surge line hot leg nozzle and charging nozzle exceeded the design code allowable of 1.0. For these two locations, BVPS will implement one or more of the following: • Further refinement of the fatigue analyses to lower the predicted CUFs to less than 1.0; • Management of fatigue at the affected locations by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method acceptable to the NRC); or • Repair or replacement of the affected locations.	January 29, 2016	LRA	A.2.3.3.2 4.3.3.3.3
26	Evaluate Unit 1 Extended Power Uprate operating experience prior to the period of extended operation for license renewal aging management program adjustments.	January 29, 2016	None	Appendix B.2

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A.5 Unit 2 License Renewal Commitments

Table A.5-1 identifies those actions committed to by FENOC for BVPS Unit 2 in the BVPS License Renewal Application (LRA). These regulatory commitments will be tracked within the FENOC regulatory commitment management program. Any other actions discussed in the LRA represent intended or planned actions by FENOC. These other actions are described only as information and are not regulatory commitments. This list will be revised as necessary in subsequent amendments to reflect changes resulting from NRC audit questions and BVPS responses to NRC requests for additional information.

Table A.5-1
Unit 2 License Renewal Commitments

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
1	Implement the Buried Piping and Tanks Inspection Program as described in LRA Section B.2.8.	Will be implemented within the 10 years prior to May 27, 2027	LRA	A.1.8 B.2.8
2	Enhance the Closed-Cycle Cooling Water System Program to: • Add the diesel-driven fire pump (Unit 1 only) and the diesel-driven standby air compressor (Unit 2 only) to the program; • Detail performance testing of heat exchangers and pumps, and provide direction to perform visual inspections of system components; • Identify closed-cycle cooling water system parameters that will be trended to determine if heat exchanger tube fouling or corrosion product buildup exists; • Control performance tests and perform visual inspections at the required frequency.	May 27, 2027	LRA	A.1.9 B.2.9

Table A.5-1
Unit 2 License Renewal Commitments
(continued)

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
3	Implement the Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements One-Time Inspection Program as described in LRA Section B.2.10.	Will be implemented within the 10 years prior to May 27, 2027	LRA	A.1.10 B.2.10
4	Implement the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program as described in LRA Section B.2.11.	May 27, 2027	LRA	A.1.11 B.2.11
5	Implement the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program as described in LRA Section B.2.12.	May 27, 2027	LRA	A.1.12 B.2.12
6	Implement the Electrical Wooden Poles/Structures Inspection Program as described in LRA Section B.2.13.	Will be implemented within the 5 years prior to May 27, 2027	LRA	A.1.13 B.2.13
7	Implement the External Surfaces Monitoring Program as described in LRA Section B.2.15.	May 27, 2027	LRA	A.1.15 B.2.15
8	Enhance the Fire Protection Program to: • Include a new attachment to the BVPS Fire Protection Program administrative procedure to address the Fire Protection Systems that are in scope for license renewal purposes;	May 27, 2027	LRA	A.1.16 B.2.16

Table A.5-1
Unit 2 License Renewal Commitments
(continued)

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
8, cont.	• Provide details of the NUREG-1801 inspection and testing guidelines, the plant implementation strategy, surveillance test and inspection frequencies, and affected implementing procedure(s); and, • Provide inspection guidance details to include degradation such as concrete cracking and spalling, and loss of material of fire barrier walls, ceilings and floors that may affect the fire rating of the assembly or barrier.			
9	Enhance the Fire Water System Program to: • Include a program requirement to perform flow test or inspection of all accessible fire water headers and piping during the period of extended operation at an interval determined by the Fire Protection System Engineer; • Include a program requirement that requires a representative number of fire water piping locations be identified if piping visual inspections are used as an alternative to non-intrusive testing; • Include a program requirement that allows test or inspection results from an accessible section of pipe to be extrapolated to an inaccessible, but similar section of pipe. If no similar section of accessible pipe is available, then alternative testing or inspection activities must be used;	Will be implemented within the 10 years prior to May 27, 2027	LRA	A.1.17 B.2.17

Table A.5-1
Unit 2 License Renewal Commitments
(continued)

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
9, cont.	• Include a program requirement that, at least once prior to the period of extended operation, all accessible Fire Protection headers and piping shall be flow tested in accordance with NFPA 25 or visually/ultrasonically inspected; • Include steps in the program procedure that require testing or replacement of sprinkler heads that will have been in service for 50 years; and, • Include a program requirement to perform a fire water subsystem internal inspection any time a subsystem (including fire pumps) is breached for repair or maintenance.			
10	Enhance the Flux Thimble Tube Inspection Program to: • Include a requirement in the program procedure to state that, if a flux thimble tube cannot be inspected over the tube length (tube length that is subject to wear due to restriction or other defect), and cannot be shown by analysis to be satisfactory for continued service, the thimble tube must be removed from service to ensure the integrity of the Reactor Coolant System pressure boundary.	May 27, 2027	LRA	A.1.19 B.2.19

**Table A.5-1
Unit 2 License Renewal Commitments
(continued)**

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
11	Enhance the Fuel Oil Chemistry Program to: • Revise the implementing procedure for sampling and testing the diesel-driven fire pump fuel oil storage tank (Unit 1 only) to include a test for particulate and accumulated water in addition to the test for sediment and water; and, • Generate a new implementing procedure for sampling and testing the security diesel generator fuel oil day tank (Common) for accumulated water, particulate contamination, and sediment / water.	May 27, 2027	LRA	A.1.20 B.2.20
12	Implement the Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program as described in LRA Section B.2.21.	May 27, 2027	LRA	A.1.21 B.2.21
13	Implement the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program as described in LRA Section B.2.22.	May 27, 2027	LRA	A.1.22 B.2.22
14	Enhance the Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems Program to: • Include guidance in the program administrative procedure to inspect for loss of material due to corrosion on Unit 2 crane and trolley structural components and rails; and,	May 27, 2027	LRA	A.1.23 B.2.23

Table A.5-1
Unit 2 License Renewal Commitments
(continued)

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
14, cont.	Include guidance in the crane and hoist inspection procedures to inspect for loss of material due to corrosion on Unit 2 crane and trolley structural components and rails or extendable arms, as appropriate.			
15	Enhance the Masonry Wall Program to: • Include in program scope additional masonry walls identified as having aging effects requiring management for license renewal.	May 27, 2027	LRA	A.1.25 B.2.25
16	Implement the Metal Enclosed Bus Program as described in LRA Section B.2.26.	May 27, 2027	LRA	A.1.26 B.2.26
17	For the Nickel-Alloy Nozzles and Penetrations Program, regarding activities for managing the aging of nickel-alloy components and nickel-alloy clad components susceptible to primary water stress corrosion cracking - PWSCC (other than upper reactor vessel closure head nozzles and penetrations), BVPS commits to develop a plant-specific aging management program that will implement applicable: 1. NRC Orders, Bulletins and Generic Letters; and, 2. Staff-accepted industry guidelines.	May 27, 2027	LRA	A.1.28 B.2.28
18	Implement the One-Time Inspection Program as described in LRA Section B.2.30.	Will be implemented within the 10 years prior to May 27, 2027	LRA	A.1.30 B.2.30

Table A.5-1
Unit 2 License Renewal Commitments
(continued)

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
19	Implement the One-Time Inspection of ASME Code Class 1 Small-Bore Piping Program as described in LRA Section B.2.31.	Will be implemented within the 10 years prior to May 27, 2027	LRA	A.1.31 B.2.31
20	For the PWR Vessel Internals Program, regarding activities for managing the aging of Reactor Vessel internal components and structures, BVPS commits to: 1. Participate in the industry programs applicable to BVPS Unit 2 for investigating and managing aging effects on reactor internals; 2. Evaluate and implement the results of the industry programs as applicable to the BVPS Unit 2 reactor internals; and, 3. Upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for the BVPS Unit 2 reactor internals to the NRC for review and approval.	May 27, 2025	LRA	A.1.33 B.2.33
21	Implement the Selective Leaching of Materials Program as described in LRA Section B.2.36.	May 27, 2027	LRA	A.1.36 B.2.36
22	Enhance the Structures Monitoring Program to: • Include in program scope additional structures and structural components identified as having aging effects requiring management for license renewal;	May 27, 2027	LRA	A.1.39 B.2.39

Table A.5-1
Unit 2 License Renewal Commitments
(continued)

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
22, cont.	• Include inspection guidance in program implementing procedures to detect significant cracking in concrete surrounding the anchors of vibrating equipment; • Include a requirement in program procedures to perform opportunistic inspections of normally inaccessible below-grade concrete when excavation work uncovers a significant depth; • Include a requirement in program procedures to perform periodic sampling of groundwater for pH, chloride concentration, and sulfate concentration; and, • Include a requirement in program procedures to monitor elastomeric materials used in seals and sealants, including compressible joints and seals, waterproofing membranes, etc., associated with in-scope structures and structural components for cracking and change in material properties.			
23	Implement the Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program as described in LRA Section B.2.40.	May 27, 2027	LRA	A.1.40 B.2.40
24	Implement the Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program as described in LRA Section B.2.41.	May 27, 2027	LRA	A.1.41 B.2.41

Table A.5-1
Unit 2 License Renewal Commitments
(continued)

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
25	Enhance the Water Chemistry Program to: • Change BVPS frequency for reactor coolant silica monitoring to once per week for Operational Modes 1 and 2, and once per day during heatup in Operational Modes 3 and 4, to be consistent with EPRI guidelines.	May 27, 2027	LRA	A.1.42 B.2.42
26	The Unit 2 steam generator secondary manway bolts and the steam generator tubes fatigue analyses are based on a 40-year life (current operating license expires in 2027). As part of the Steam Generator Tube Integrity Program, BVPS will perform a reanalysis, repair, or replacement of the affected components such that the design basis of these components is not exceeded for the period of extended operation.	May 27, 2027	LRA	A.3.3.1 4.3.1
27	BVPS will perform an assessment of the Unit 2 Emergency Diesel Generator Air Start System to determine whether the full-temperature cycles limit would be exceeded for 60 years of operation. This assessment will be performed prior to the period of extended operation.	May 27, 2027	LRA	A.3.3.2.1 4.3.2.1

Table A.5-1
Unit 2 License Renewal Commitments
(continued)

Item Number	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
28	Of the NUREG/CR-6260 locations, the U_{env} (60 years) of the Unit 2 surge line nozzle, charging nozzle, and RHR line exceeded the design code allowable of 1.0. For these three locations, BVPS will implement one or more of the following: • Further refinement of the fatigue analyses to lower the predicted CUFs to less than 1.0; • Management of fatigue at the affected locations by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method acceptable to the NRC); or, • Repair or replacement of the affected locations.	May 27, 2027	LRA	A.3.3.3.3 4.3.3.3.3
29	Evaluate Unit 2 Extended Power Uprate operating experience prior to the period of extended operation for license renewal aging management program adjustments.	May 27, 2027	None	Appendix B.2