CHAPTER 19

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CHAPTER 19

MANAGING THE EFFECTS OF COMPONENT AGING

19.0 INTRODUCTION

This section provides a summary description of the programs and activities for managing the effects of aging and the evaluation of time-limited aging analyses in accordance with 10 CFR 54.21(d). These programs and activities were developed to support renewal of the original operating license for Beaver Valley Power Station Unit No. 2 that was scheduled to expire on May 27, 2027.

An integrated plant assessment in support of license renewal identified the aging management programs and activities necessary to provide reasonable assurance that components within the scope of license renewal will continue to perform their intended functions for the period of extended operation. The period of extended operation is the 20 year period ending May 27, 2047.

For each of the plant-specific time-limited aging analyses, the evaluations have determined that the analyses remain valid for the period of extended operation, the analyses have been projected to the end of the period of extended operation, or that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Appendix A of both NUREG-1929, ''Safety Evaluation Report Related to the License Renewal of the Beaver Valley Power Station, Units 1 and 2,'' and Supplement 1 to NUREG-1929 (both published in October 2009), identified commitments associated with the aging management programs and activities to manage aging effects for structures and components. These commitments are provided in Table 19-1, Unit 2 License Renewal Commitments.

19.1 SUMMARY DESCRIPTIONS OF AGING MANAGEMENT PROGRAMS

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.

19.1.1 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 NEI 94-01, *Industry Guidance for Implementing Performance-Based Options of 10 CFR Part 50 Appendix J* [Reference 19.1-2], with conditions and limitations specified in NEI 94-01, Revision 2-A.

19.1.2 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 19.1-42) where applicable.

19.1.3 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.

19.1.4 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 19.1-5].

19.1.5 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. 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.

19.1.6 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 19.1-6], and EPRI NP-5769, *Degradation and Failure of Bolting in Nuclear Power Plants* [Reference 19.1-7]. Also, it implements industry recommendations for comprehensive bolting maintenance, as delineated in EPRI TR-104213, *Bolted Joint Maintenance & Application Guide* [Reference 19.1-8], 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

19.1.7 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 19.1-9].

The Boric Acid Corrosion Control Program ensures that the pressure boundary integrity and the material condition of structures, systems, and components in contact with evidence of borated water leakage are maintained consistent with the current licensing basis during the period of extended operation. The Boric Acid Corrosion Control Program includes:

a) Visual inspection of external surfaces that are potentially exposed to borated water leakage

b) Timely discovery of the leak path

c) Initiation of appropriate corrective action

d) Assessment of the damage (if identified)

e) Follow-up inspection for adequacy of corrective action

19.1.8 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.

19.1.9 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 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 19.1-33] 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. Water chemistry is monitored and controlled using procedures and processes that implement EPRI closed cooling water chemistry guidelines with certain differences as described in the BVPS closed cycle cooling water chemistry program document.

19.1.10 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 tested. Electrical connections covered under the Environmental Qualification (EQ) Program (Section 19.1-14), 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 confirm that the loosening of cable 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 voltage level (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.

The metallic parts of metal enclosed bus connections are managed by the Metal Enclosed Bus Program (Unit 2 only) (Section 19.1-26), and are therefore not included within the scope of the program.

19.1.11 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.

19.1.12 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.

19.1.13 Electrical Wooden Poles/Structures Inspection Program

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.

19.1.14 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.

EQ Component Reanalysis Attributes

The reanalysis of an aging evaluation is normally performed to extend the qualification by reducing excess conservatism incorporated in the prior evaluation. Reanalysis of an aging evaluation to extend the qualification of a component is performed on a routine basis pursuant to 10 CFR 50.49(e) as part of the BVPS EQ Program. While a component life-limiting condition may be due to thermal, radiation or cyclical aging, the vast majority of component aging limits are based on thermal conditions. Conservatism may exist in aging evaluation parameters, such as the assumed ambient temperature of the component, an unrealistically low activation energy, or in the application of a component (de-energized versus energized). The reanalysis of an aging evaluation is documented according to BVPS quality assurance program requirements, which require the verification of assumptions and conclusions. Important attributes of a reanalysis include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, and corrective actions (if acceptance criteria are not met). These attributes are discussed in the following four subsections.

- **Analytical Methods:** The analytical models used in the reanalysis of an aging evaluation are the same as those previously applied during the original evaluation. The Arrhenius methodology is an acceptable model for a thermal aging evaluation. For license renewal radiation aging evaluation, 60-year normal radiation dose is established by extrapolating the 40-year normal dose (40 year dose X 1.5) plus accident radiation dose. 60-year cyclical aging is established in a similar manner. Other models may be justified on a case-by-case basis.

- **Data Collection and Reduction Methods:** Reducing excess conservatism in the component service conditions (for example, temperature, radiation, and cycles) used in the prior aging evaluation is the chief method used for a reanalysis. Actual monitored service conditions, such as temperature, are typically lower than the design service conditions used in the prior aging evaluation and, therefore, can support extended thermal life of the equipment.

- **Underlying Assumptions:** EQ component aging evaluations contain sufficient conservatism to account for most environmental changes occurring due to plant modifications and events. When unexpected adverse conditions are identified during operational or maintenance activities that affect the normal operating environment of a qualified component, the affected EQ component is evaluated and appropriate corrective actions are taken, which may include changes to the qualification bases and conclusions. Excess conservatism in thermal life analysis may be reduced by reevaluating material activation energy, to justify a higher value that would support extended life at elevated temperature. Similar methods of reducing excess conservatism in the component service conditions and material properties used in prior aging evaluations may be used for radiation and cyclical aging. Any changes to material activation energy will be justified.

- **Acceptance Criteria and Corrective Actions:** If qualification cannot be extended by reanalysis, the component is refurbished or replaced prior to exceeding the period for which the current qualification remains valid. A reanalysis is to be performed in a timely manner (that is, sufficient time is available to refurbish, replace or requalify the component if reanalysis is unsuccessful).

The Environmental Qualification (EQ) of Electric Components Program is an existing program established to meet BVPS commitments for 10 CFR 50.49. It is consistent with NUREG-1801 [Reference 19.1-33], Section X.E1, *Environmental Qualification (EQ) of Electric Components*.

This program includes consideration of operating experience to modify qualification bases and conclusions, including qualified life. Compliance with 10 CFR 50.49 provides reasonable assurance that components can perform their intended function(s) during accident conditions after experiencing the effects of inservice aging. Consistent with NRC guidance provided in RIS 2003-09, *Environmental Qualification of Low-Voltage Instrumentation and Control Cables*, no additional information is required to address GSI 168, *Environmental Qualification of Low-Voltage Instrumentation and Control (I&C) Cables* [Reference 19.1-34].

19.1.15 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 radiators (fins and tubes) associated with diesel engines and diesel-driven equipment 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.

19.1.16 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, as specified in program administrative procedures. 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 through periodic visual inspections. It also manages the aging effects on the diesel engine-driven fire pump fuel oil supply line through operational testing of the pump, which confirms that the component intended function is maintained. The Fire Protection Program also manages the aging effects on the halon and carbon dioxide fire suppression systems through periodic inspection and functional testing.

19.1.17 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 19.1-11]. 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.

19.1.18 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 19.1-12]. 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 has been expanded to manage wall thinning in components due to erosion mechanisms such as cavitation, flashing, droplet impingement and solid particle impingement. 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.

19.1.19 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 19.1-13].

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.

19.1.20 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 19.1-30) will be used to verify the effectiveness of the Fuel Oil Chemistry Program.

19.1.21 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 managed such 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 (2) years, to minimize cable exposure to significant moisture, such as inspecting for water collection in cable manholes, 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 will be determined prior to the initial test, and is to be a proven test for detecting deterioration of the insulation system due to wetting, such as power factor, partial discharge, or other testing that is state-of-the-art at the time the test is performed. Testing is conducted at least once every ten (10) years, with initial testing completed prior to the period of extended operation. Also, periodic visual inspections are performed on the accessible portions of cables (i.e., in manholes) for water induced damage. These inspections are performed at least once every two (2) years, with the first inspection completed prior to the period of extended operation.

19.1.22 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.

19.1.23 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 19.1-14].

19.1.24 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 19.1-30) will be used to verify the effectiveness of the Lubricating Oil Analysis Program.

19.1.25 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 19.1-15], and NRC Information Notice 87-67, *Lessons Learned from Regional Inspections of Licensee Actions in Response to IE Bulletin 80-11* [Reference 19.1-16]. 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 19.1-39) 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.

19.1.26 Metal Enclosed Bus Program

The Metal Enclosed Bus Program is applicable to the isolated phase bus at both units, and 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. The bolted connections for the isolated phase bus are checked by using thermography or by measuring connection resistance. Inspections are completed prior to the period of extended operation and every 10 years thereafter.

19.1.27 Metal Fatigue of Reactor Coolant Pressure Boundary Program

Program Description

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 RCS components to prevent the fatigue design limit from being exceeded. 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 19.1-17], and the guidance of EPRI Technical Report MRP-47, *Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application* [Reference 19.1-18]. 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 19.1-19], and in NUREG/CR 5704, *Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels* [Reference 19.1-20].

Aging Management Program Elements

The results of an evaluation of each of the 10 aging management program elements described in NUREG-1801 [Reference 19.1-33], Section X.M1, are provided as follows:

- Scope of Program

  The program tracks critical transient cycles to ensure RCS components remain within their design fatigue usage limits. This program utilizes the systematic counting of operational cycles to ensure that component design fatigue usage limits are not exceeded.

  Fatigue analyses validated by this program include nuclear steam supply system (NSSS) equipment as well as the ASME Class 1 portions of the primary piping.

  The program addresses the effects of the reactor coolant environment on component fatigue life by including, within the program scope, environmental fatigue evaluations of the sample locations specified in NUREG/CR-6260.

- Preventive Actions

  The program provides for monitoring and tracking critical thermal and pressure transients for RCS 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 unit. These critical transients include plant heatup, plant cooldown, inadvertent auxiliary spray, and RCS cold overpressurization. Supplemental transients were also identified by the program for monitoring. These supplemental transients include pressurizer insurge transient, selected Chemical and Volume Control System transients, Auxiliary Feedwater injections and RHR actuation.

  The number of critical transient occurrences is periodically reviewed (annual basis) to determine if there are any adverse trends; adverse conditions or deficient conditions with the primary objective of initiating evaluation of adverse trends and adverse conditions early to prevent the possibility of a deficient condition. Adverse Trend is an observed increase in the rate of critical transient occurrences that, if it continued, would result in exceeding the fatigue cycle design limit number of transients prior to the end of the Unit's 60-year operating life. Adverse Condition is a condition in which the number of actual transient occurrences exceeds 80% of the fatigue cycle design limit number of occurrences. Deficient Condition is a condition in which the current number of actual transient occurrences exceeds the fatigue cycle design limit number of occurrences. Adverse trends and adverse conditions are evaluated by Engineering to determine if and when more rigorous analysis or alternate resolutions are required. Deficient conditions are addressed under the BVPS Corrective Action Program.

- Parameters Monitored / Inspected

The design of ASME Class 1 and other specific components considered a predicted number of fatigue cycles for various design transients. Monitoring the actual number of transient occurrences serves to confirm the adequacy of the design analysis. The program monitors and tracks fatigue significant temperature and pressure transients in order not to exceed the design limit on fatigue usage.

The program, for the most part, is a transient cycle counting program and does not require analysis of operational data (local monitoring) to obtain an effective number of transients.

The WESTEMS™ Integrated Diagnostics and Monitoring System analysis for the surge line to hot leg nozzles is based on past occurrences of various transients along with what are believed to be conservative assumptions of future transients. In this case, the input assumptions will be verified by periodic reanalysis using updated plant history files.

- Detection of Aging Effects

The program requires the systematic counting of operational cycles to ensure that component design fatigue usage limits are not exceeded. When the accrued operational cycles approach the component design cycles, corrective action is required by the program to ensure the design cycle limit is not exceeded. If the corrective action has an impact on the cumulative fatigue usage factor (CUF), an updated CUF will be generated.

- Monitoring and Trending

The design of ASME Class 1 and other specific components considered a predicted number of fatigue cycles for various design transients. Monitoring the actual number of transient occurrences serves to confirm the adequacy of the design analysis. The program monitors and tracks fatigue significant temperature and pressure transients in order not to exceed the design limit on fatigue usage.

- Acceptance Criteria

  The program verifies that the fatigue usage remains below the design code limit considering environmental fatigue effects as described under the program description.

- Operating Experience

  Concerns for the overall health of the transient/cycle counting program were documented using the FENOC Corrective Action Program. Corrective actions included identifying a program owner, developing an administration program document and updating it to incorporate responsibilities, improving cycle counting, and establishing a process for engineering to evaluate plant data. Fatigue monitoring to date indicates that the number of design transient events assumed in the original design analysis will be sufficient for a 60-year operating period. The program has remained responsive to emerging issues and concerns, particularly the pressurizer surge and spray nozzle, hot leg surge nozzle, and surge line transients.

  For example, in 2002, a Westinghouse evaluation identified that the BVPS Unit 2 letdown, charging, and excess letdown piping could potentially exceed their design allowable cycle counts for several design transients. However, further evaluation of existing plant operations and the physical separation distance of the letdown and excess letdown piping demonstrated that no further evaluation of the letdown or excess letdown piping was required for current operation or for the period of extended operation. A re-analysis of the charging piping was required to account for the appropriate transients for a 60-year plant life.

  This responsiveness to emerging issues and continued program improvements provide evidence that the program will remain effective for managing cumulative fatigue damage for passive components.

Conclusion

Continued implementation of the Metal Fatigue of Reactor Coolant Pressure Boundary Program provides reasonable assurance that the aging effects will be managed so that the systems and components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

19.1.28 [Deleted]

19.1.29 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 19.1-21], 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 19.1-22]. Detection of cracking is accomplished through implementation of a combination of bare metal visual examination (external surface of head) and non-visual examination techniques.

19.1.30 One-Time Inspection Program

The One-Time Inspection Program requires one-time inspections to verify effectiveness of the Water Chemistry Program (Section 19.1-42), the Fuel Oil Chemistry Program (Section 19.1-20), and the Lubricating Oil Analysis Program (Section 19.1-24). 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 aging effects are not occurring in the following components:

- Loss of material of the steam generator feedwater ring; and,
- Loss of material of 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.

19.1.31 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 and systems less than 4 inches nominal pipe size (less than NPS 4) and greater than or equal to NPS 1, which includes pipes, fittings, branch connections, and full and partial penetration (socket) welds. The program will manage this aging effect by performing volumetric examinations for selected ASME Code Class 1 small-bore 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.

19.1.32 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 19.1-23], 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 Service Water System or structures and components serviced by the system.

19.1.33 [Deleted]

19.1.34 Reactor Head Closure Studs Program

The Reactor Head Closure Studs Program 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 19.1-24].

19.1.35 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 19.1-25] (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). 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 reference temperature for pressurized thermal shock (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. Standby capsules are installed at Unit 1 and Unit 2 and are available to provide neutron fluence monitoring and meaningful metallurgical test data for 60 and 80 years of operation. In the case where the reactor vessel has all surveillance capsules removed, the program requires use of alternative dosimetry (ex-vessel neutron dosimetry) to monitor neutron fluence during the period of extended operation.

19.1.36 Selective Leaching of Materials Inspection Program

When performing the operating experience review for License Renewal, selective leaching was identified in buried gray cast iron fire protection piping. Therefore, for this set of components, the program performs periodic testing to ensure that selective leaching is identified before a loss of component intended function.

For the remainder of the components that credit the program, a one-time visual inspection and hardness examination of selected components that are susceptible to selective leaching is performed. This 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% and gray cast iron exposed to a raw water, treated water, air, or condensation environment.

Should evidence of significant aging be revealed by the one-time inspection, the corrective action process is used for the unacceptable inspection findings. The resolution evaluates expansion of the inspection sample size, locations, and frequency.

19.1.37 Settlement Monitoring Program

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 19.2.6.3. 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.

19.1.38 Steam Generator Tube Integrity Program

The Steam Generator Tube Integrity Program is based on NEI 97-06, *Steam Generator Program Guidelines* [Reference 19.1-27]. 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.

19.1.39 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 19.1-28] and Regulatory Guide 1.160, *Monitoring the Effectiveness of Maintenance at Nuclear Power Plants* [Reference 19.1-29].

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.

19.1.40 [Deleted]

19.1.41 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 19.1-30], 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 250° C (> 482° 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.

19.1.42 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. Water chemistry is monitored and controlled using procedures and processes that implement EPRI primary and secondary water chemistry guidelines with certain differences as described in BVPS primary and secondary water chemistry program documents.

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 [Reference 19.1-33] that require augmentation of the Water Chemistry Program.

19.1.43 Not Applicable (Unit 1 Only)

19.1.44 References

19.1-1 Deleted

19.1-2 NEI 94-01, *Industry Guidance for Implementing Performance-Based Options of 10 CFR Part 50 Appendix J*, Rev. 3-A.

19.1-3 Deleted

19.1-4 Deleted

19.1-5 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.

19.1-6 NUREG-1339, *Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants*, October 17, 1991.

19.1-7 EPRI NP-5769, *Degradation and Failure of Bolting in Nuclear Power Plants*, May 5, 1988.

19.1-8 EPRI TR-104213, *Bolted Joint Maintenance & Application Guide*, December 1, 1995.

19.1-9 NRC Generic Letter 88-05, *Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants*, March 17, 1988.

19.1-10 Deleted

19.1-11 National Fire Protection Association NFPA 25, *Standard for the Inspection, Testing and Maintenance of Water-Based Fire Protection Systems*, 2002 Edition.

19.1-12 NSAC-202L-R2, *Recommendations for an Effective Flow Accelerated Corrosion Program*, April 1999.

19.1-13 NRC IE Bulletin 88-09, *Thimble Tube Thinning in Westinghouse Reactors,* July 26, 1988.

19.1-14 NUREG-0612, *Control of Heavy Loads at Nuclear Power Plants*, July 1980.

19.1-15 NRC IE Bulletin 80-11, *Masonry Wall Design*, May 8, 1980.

19.1-16 NRC Information Notice 87-67, *Lessons Learned from Regional Inspections of Licensee Actions in Response to IE Bulletin 80-11*, December 31, 1987.

19.1-17 NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves for Selected Nuclear Power Plant Components*, February 28, 1995.

19.1-18 EPRI Technical Report MRP-47, *Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application*, September 1, 2005.

19.1-19 NUREG/CR-6583, *Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low Alloy Steels*, February 1998.

19.1-20 NUREG/CR-5704, *Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels*, April 1999.

19.1-21 NRC Order EA 03-009, *Issuance of Order Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors*, February 11, 2003.

19.1-22 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.

19.1-23 NRC Generic Letter 89-13, *Service Water System Problems Affecting Safety-Related Equipment*, including Supplement 1, July 18, 1989.

19.1-24 Regulatory Guide 1.65, *Materials and Inspections for Reactor Vessel Closure Studs*, October 1973.

19.1-25 ASTM Standard E 185-82, *Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels*, June 2002.

19.1-26 Deleted

19.1-27 NEI 97-06, *Steam Generator Program Guidelines*, Rev. 2, May 2005.

19.1-28 NUMARC 93-01, *Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*, Rev. 3, October 8, 1999.

19.1-29 Regulatory Guide 1.160, *Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*, Rev. 2, March 1997.

19.1-30 ASME Code Case N-481, *Alternate Examination Requirements for Cast Austenitic Pump Casings*, May 20, 1998.

19.1-31 Deleted

19.1-32 Deleted

19.1-33 NUREG-1801, *Generic Aging Lessons Learned (GALL) Report*, Rev. 1, September 2005.

19.1-34 NRC Regulatory Issue Summary 2003-09, *Environmental Qualification of Low-Voltage Instrumentation and Control Cables*, May 2, 2003.

19.2 EVALUATION SUMMARIES OF UNIT 2 TIME-LIMITED AGING ANALYSES

19.2.1 Introduction

Time-limited aging analyses (TLAAs) are defined in 10 CFR 54.3 [Reference 19.2-3] 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 19.2.2)

- Metal Fatigue (Section 19.2.3)

- Environmental Qualification (EQ) of Electric Equipment (Section 19.2.4)

- Containment Liner Plate, Metal Containment, and Penetrations Fatigue (Section 19.2.5)

- Other Plant-Specific Time-Limited Aging Analyses (Section 19.2.6)

- References (Section 19.2.7)

19.2.2 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 19.2.2.1)

- Pressurized Thermal Shock (Section 19.2.2.2)

- Charpy Upper Shelf Energy (Section 19.2.2.3)

- Pressure-Temperature Limits (Section 19.2.2.4)

19.2.2.1 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 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 19.2-4]. For the 60-year license, WCAP-16527-NP Supplement 1, *Analysis of Capsule X from First Energy Company Beaver Valley Unit 2 Reactor Vessel Radiation Surveillance Program* [Reference 19.2-5] 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 X analysis, and comply with Reg. Guide 1.190, *Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence* [Reference 19.2-6].

The fluence projections include fuel cycle-specific calculated neutron exposures at the end of Cycle 14 (Fall 2009) as well as future projections to the end of Cycle 15 (Spring 2011) and for several intervals extending to 60 effective full power years (EFPY). The calculations account for a core power uprate to 2900 megawatts-thermal (MWt) at the onset of Cycle 14. Neutron exposure projections beyond the end of Cycle 15 were based on the spatial power distributions and associated plant characteristics of Cycles 14 and 15 in conjunction with the uprated power level.

19.2.2.2 Pressurized Thermal Shock

In the spring of 2005, Surveillance Capsule X was pulled and the analysis was documented in WCAP-16527-NP [Reference 19.2-4]. For the 60-year license, WCAP-16527-NP Supplement 1 [Reference 19.2-5] documents the EOLE analysis for pressurized thermal shock (PTS).

Using the prescribed PTS Rule (10 CFR 50.61 [Reference 19.2-7]) methodology, reference temperature for pressurized thermal shock (RT_{PTS}) values were generated for beltline and extended beltline region materials of the BVPS Unit 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).

19.2.2.3 Charpy Upper Shelf Energy

In the spring of 2005, Surveillance Capsule X was pulled and the analysis was documented in WCAP-16527-NP [Reference 19.2-4]. For the 60-year license, WCAP-16527-NP Supplement 1 [Reference 19.2-5] 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 19.2-8], 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 2 C_VUSE analysis has been dispositioned in accordance with 10 CFR 54.21(c)(1)(ii).

19.2.2.4 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 19.2-7], 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 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.

19.2.3 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 19.2.3.1)
- Non-Class 1 Fatigue Evaluations (Section 19.2.3.2)
- Generic Industry Issues on Fatigue (Section 19.2.3.3)

19.2.3.1 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 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).

19.2.3.1.1 Unit 2 Charging Line

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 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 Unit 2 charging line is not exceeded for the period of extended operation. Therefore, the charging line fatigue TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

19.2.3.1.2 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. The Metal Fatigue of Reactor Coolant Pressure Boundary Program will be enhanced to include a requirement that provides for reanalysis, repair, or replacement of the Unit 2 steam generator secondary manway bolts and the steam generator tubes such that the design bases of 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).

19.2.3.1.3 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, FENOC decided to further revise the operating procedures to optimize the plant shutdown and startup processes for Unit 2. The FENOC 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 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).

19.2.3.2 Non-Class 1 Fatigue Evaluations

19.2.3.2.1 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 (SA).

For those non-Class 1 components identified as subject to cracking due to fatigue, a review of system operating characteristics was conducted by FENOC 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.

FENOC 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 Unit 2 EDG Air Start system is designed as a stand-by system. The compressor runs to fill the air start tank and does not run again until the tank needs to be either topped off or refilled after discharge. The act of compressing the air is the only source of heat for this piping. Measurement of discharge temperature during compressor operation indicates that the compressors, bolting, discharge piping and valves may exceed the threshold temperature for thermal fatigue during compressor operation. Confirmation of the full-temperature cycles of the Unit 2 EDG Air Start System would be a time consuming process and would not account for variables such as the leak tightness of the air system. Therefore, the design analysis for the piping has been revised to incorporate the observed temperatures as a new load case. The stress levels for this thermal load case are below the endurance limit for the piping material. In other words, the revised analysis has qualified the air start piping for an infinite number of thermal cycles at the observed temperatures. Therefore, the Unit 2 EDG Air Start System fatigue TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(ii).

19.2.3.2.2 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, cumulative fatigue damage 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.

For non-Class 1 pressure vessels, heat exchangers, storage tanks, pumps, and turbine casings identified as subject to cracking due to fatigue, a review of system operating characteristics was conducted by FENOC 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 original design (usually 7,000 cycles, as described in Section 19.2.3.2.1), 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. FENOC evaluated the validity of this assumption for 60 years of plant operation. 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 pressure vessels, heat exchangers, storage tanks, pumps, and turbine casings fatigue TLAAs have been dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

19.2.3.3 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.

19.2.3.3.1 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 19.2-10], 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).

19.2.3.3.2 Pressurizer Surge Line Thermal Stratification (NRC Bulletin 88-11)

NRC Bulletin 88-11, *Pressurizer Surge Line Thermal Stratification* [Reference 19.2-11], 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 19.2-12], 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 19.2-13] 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. Since 60-year projected operational cycles were used in determining that the 200 heatup and cooldown transient assumption remains bounding for the period of extended operation, the Metal Fatigue of Reactor Coolant Pressure Boundary Program must continue to be used to validate this assumption. Therefore, the Unit 2 pressurizer surge line thermal stratification TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

19.2.3.3.3 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 19.2-14], 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 19.2-15]. 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 19.2-16], and NUREG/CR-5704, *Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels* [Reference 19.2-17]. 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 the pressurizer surge line to hot leg nozzle, U_{env} exceeded the design code allowable limit of 1.0. For this location, 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.

The U_{env} at the other NUREG/CR-6260 locations (reactor vessel shell and lower head, reactor vessel inlet and outlet nozzles, charging system nozzle, safety injection system nozzle and RHR system piping), have been demonstrated to remain less than the design code allowable limit of 1.0 for the period of extended operation.

As discussed in Section 19.2.3.1, since 60-year projected operational cycles were used in determining that the design fatigue analyses remain valid for the period of extended operation, the Metal Fatigue of Reactor Coolant Pressure Boundary Program must continue to be used to validate the assumptions used in the evaluations. Therefore, the TLAAs associated with the NUREG/CR-6260 locations have been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

19.2.4 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).

19.2.5 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 19.2.5.1)
- Containment Liner Corrosion Allowance (Section 19.2.5.2)
- Containment Liner Penetration Fatigue (Section 19.2.5.3)

19.2.5.1 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).

19.2.5.2 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 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 19.2-18]. 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 19.2-19]. 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).

19.2.5.3 Containment Liner Penetration Fatigue

19.2.5.3.1 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).

19.2.5.3.2 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).

19.2.5.3.3 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).

19.2.6 Other Plant-Specific Time-Limited Aging Analyses

The plant-specific TLAAs summarized in this section include:

- Leak Before Break (Section 19.2.6.1)
- High Energy Line Break Postulation (Section 19.2.6.2)
- Settlement Of Structures (Section 19.2.6.3)
- Crane Load Cycles (Section 19.2.6.4)

19.2.6.1 Leak Before Break

Leak before break (LBB) analyses evaluate postulated flaw growth in piping to alter the structural design basis. FENOC 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. FENOC 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).

19.2.6.1.1 Main Coolant Loop Piping Leak Before Break

The LBB evaluation for the main reactor coolant loop piping was 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 19.2-20]. This evaluation was approved by the NRC in an SER [Reference 19.2-21] in 1991.

Supplement 1 to WCAP-11923 [Reference 19.2-31] was issued in 2007 to incorporate the Power Uprate and License Renewal projects.

The reactor coolant loop piping was reanalyzed for LBB in WCAP-17488-P [Reference 19.2-32] to incorporate the latest piping loads, operating conditions and Standard Review Plan criteria.

19.2.6.1.2 Pressurizer Surge Line Piping Leak Before Break

The LBB evaluation for the pressurizer surge line piping was documented in WCAP-12093, *Evaluation of Thermal Stratification for the Beaver Valley Unit 2 Pressurizer Surge Line*. This evaluation (including Supplements 1 and 2) [Reference 19.2-12] was approved by the NRC in an SER [Reference 19.2-13] 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 19.2-12], was prepared and submitted to the NRC [Reference 19.2-24]. This evaluation was approved by the NRC in an SER [Reference 19.2-25] in 1991.

Supplements 4 and 5 to WCAP-12093 [Reference 19.2-33] were issued in 2007 to incorporate the Power Uprate and License Renewal projects.

The pressurizer surge line nozzle was impacted by the Pressurizer Weld Overlay Project. Weld overlay was performed during the Cycle 12 Refueling Outage (2006). In RIS 2010-07, [Reference 19.2-34], the NRC expressed concern that the application of structural weld overlay may affect the results of Leak-Before-Break (LBB) analysis. The pressurizer surge line was reanalyzed for LBB in WCAP-17394-P [Reference 19.2-35] to address weld overlay and RIS 2010-07.

19.2.6.1.3 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 19.2-26].

19.2.6.2 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 19.2-27], 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 19.2-28]. 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).

19.2.6.3 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 19.1-37). 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 stress analyses are maintained for the period of extended operation. Therefore, the TLAAs associated with the piping stress analyses have been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).19.2.6.4 Crane Load Cycles

In the response to NUREG-0612, *Control of Heavy Loads at Nuclear Power Plants* [Reference 19.2-29], 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 19.2-30]. 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).

19.2.7 References

19.2-1 BVPS License Renewal Application, FENOC Letter L-07-113, August 27, 2007.

19.2-2 NRC SER for BVPS License Renewal, Volumes 1 & 2, with Supplement, ML093020276, ML093000278 and ML09250014.

19.2-3 10 CFR 54, *Requirements for Renewal of Operating Licenses for Nuclear Power Plants*.

19.2-4 WCAP-16527-NP, *Analysis of Capsule X from First Energy Nuclear Operating Company Beaver Valley Unit 2 Reactor Vessel Radiation Surveillance Program*, Rev. 0.

19.2-5 WCAP-16527-NP Supplement 1, Revision 1, *Analysis of Capsule X from First Energy Nuclear Operating Company Beaver Valley Unit 2 Reactor Vessel Radiation Surveillance Program*, September 2011.

19.2-6 Regulatory Guide 1.190, *Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence*, March 2001.

19.2-7 10 CFR 50, *Domestic Licensing of Production and Utilization Facilities*.

19.2-8 Regulatory Guide 1.99, *Radiation Embrittlement of Reactor Vessel Materials*, Rev. 2.

19.2-9 Deleted.

19.2-10 NRC Bulletin 88-08, *Thermal Stresses in Piping Connected to Reactor Coolant Systems*, June 22, 1988, (including Supplements 1 and 2).

19.2-11 NRC Bulletin 88-11, *Pressurizer Surge Line Thermal Stratification*, December 20, 1988.

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19.2-13 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.

19.2-14 NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components*, February 1995.

19.2-15 Generic Safety Issue (GSI)-190, *Fatigue Evaluation of Metal Components for 60-Year Plant Life*, Rev. 2.

19.2-16 NUREG/CR-6583, *Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low Alloy Steels*, February 1998.

19.2-17 NUREG/CR-5704, *Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels*, March 1999.

19.2-18 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.

19.2-19 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.

19.2-20 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.

19.2-21 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.

19.2-22 Deleted

19.2-23 Deleted

19.2-24 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.

19.2-25 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.

19.2-26 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.

19.2-27 Regulatory Guide 1.46, *Protection Against Pipe Whip Inside Containment*, May 1973.

19.2-28 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.

19.2-29 NUREG-0612, *Control of Heavy Loads at Nuclear Power Plants*, July 1980.

19.2-30 Crane Manufacturers Association of America Specification #70 (CMAA-70), *Specifications for Electric Overhead Traveling Cranes*, Revised 1983.

19.2-31 WCAP-11923-P, Supplement 1, *EPU/LR Project Update of Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Beaver Valley Unit 2*, March 2007.

19.2-32 WCAP-17488-P, *Leak-Before-Break Analysis Update for the Beaver Valley Unit 2 Primary Loop Piping*, February 2012.

19.2-33 WCAP-12093-P, Supplement 4, *EPU Project Update of Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for Beaver Valley Unit 2*, and Supplement 5, *EPU Project Update of Evaluation of Thermal Stratification for the Beaver Valley Unit 2 Pressurizer Surge Line*, March 2007.

19.2-34 RIS 2010-07, NRC Regulatory Issue Summary 2010-07 - *Regulatory Requirements for Application of Weld Overlays and Other Mitigation Techniques in Piping Systems Approved for Leak-Before-Break*, June 8, 2010.

19.2-35 WCAP-17394-P, *Leak-Before-Break Analysis Update for the Beaver Valley Unit 2 Pressurizer Surge Line*, December 2011.

Tables for Chapter 19

TABLE 19-1

UNIT 2 LICENSE RENEWAL COMMITMENTS

Table 19-1 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.

Table 19-1

Item No.	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 19-1, cont.

Item No.	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. Prior to implementation of the program, evaluate the program against the final approved version of NRC License Renewal Interim Staff Guidance LR-ISG-2007-02, “Changes To Generic Aging Lesson Learned (GALL) Report Aging Management Program (AMP) XI.E6, “Electrical Cable Connections Not Subject To 10 CFR 50.49 Environmental Qualification Requirements,”” when issued, and revise the program to be consistent with the NRC Interim Staff Guidance.	Will be implemented within the 10 years prior to May 27, 2027	LRA and FENOC Letter L-08-262	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

Table 19-1, cont.				
Item No.	**Commitment**	**Implementation Schedule**	**Source**	**Related LRA Section No./ Comments**
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; • Provide details of the NUREG-1801 inspection and testing guidelines, the plant implementation strategy, surveillance test and inspection frequencies (inspection frequency of the Halon and CO2 systems will be changed to at least once every 6 months), 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.	May 27, 2027	LRA and FENOC Letter L-08-375	A.1.16 B.2.16

Table 19-1, cont.				
Item No.	**Commitment**	**Implementation Schedule**	**Source**	**Related LRA Section No./ Comments**
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; • 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.	Will be implemented within the 10 years prior to May 27, 2027	LRA	A.1.17 B.2.17

Table 19-1, cont.				
Item No.	**Commitment**	**Implementation Schedule**	**Source**	**Related LRA Section No./ Comments**
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
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; • 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; and, • Revise implementing procedures to perform UT thickness measurements of accessible above-ground fuel oil tank bottoms at the same frequency as tank cleaning and inspections to ensure that significant degradation is not occurring. For inaccessible tank bottoms, determine tank bottom thickness using an appropriate NDE technique if inspections indicate the presence of significant corrosion.	May 27, 2027	LRA, FENOC Letter L-08-262 and FENOC Letter L-08-316	A.1.20 B.2.20

Table 19-1, cont.				
Item No.	**Commitment**	**Implementation Schedule**	**Source**	**Related LRA Section No./ Comments**
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. BVPS commits to implement one of the following prior to entering the period of extended operation: 1. Adopt an acceptable methodology that demonstrates that the in-scope, continuously submerged, inaccessible, medium-voltage cables will continue to perform their intended function during the period of extended operation. -or- 2. Implement measures to minimize cable exposure to significant moisture through dewatering manholes. Incorporate operating experience obtained from dewatering and adjust the dewatering frequency to minimize cable exposure to significant moisture. [Significant moisture is defined as periodic exposures to moisture that last more than a few days (e.g., cable in standing water). Periodic exposures to moisture that last less than a few days (i.e., normal rain and drain) are not significant.] -or- 3. Replace the in-scope, continuously submerged medium-voltage cables with cables designed for submerged service.	May 27, 2027	LRA; FENOC Letter L-08-262; FENOC Letter L-09-057; FENOC Letter L-09-138; and FENOC Letter L-09-151	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

Table 19-1, cont.

Item No.	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
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, • 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.	May 27, 2027	LRA	A.1.23 B.2.23
15	Enhance the Masonry Wall Program to: • Include in program scope additional masonry walls identified as having aging effects requiring management for license renewal; and, • Include a requirement in program procedures to incorporate the results of the Masonry Wall Program inspection and document the condition of the walls in the inspection report.	May 27, 2027	LRA and FENOC Letter L-08-262	A.1.25 B.2.25
16	Implement the Metal Enclosed Bus Program as described in LRA Section B.2.26, and as amended by UFSAR change notice CN 12-185.	May 27, 2027	LRA	A.1.26 B.2.26

Table 19-1, cont.				
Item No.	**Commitment**	**Implementation Schedule**	**Source**	**Related LRA Section No./ Comments**
17	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 and FENOC Letter L-08-212	None
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
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

Table 19-1, cont.				
Item No.	**Commitment**	**Implementation Schedule**	**Source**	**Related LRA Section No./ Comments**
20	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 and FENOC Letter L-08-212	None
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; • Include inspection guidance in program implementing procedures to detect significant cracking in concrete surrounding the anchors of vibrating equipment;	May 27, 2027 for all enhancements except groundwater sampling (4^{th} bullet). Groundwater sampling will be implemented five (5) years prior to entering the period of extended operation, then continue on a five (5) year interval thereafter.	LRA, FENOC Letter L-08-181 and FENOC Letter L-08-262	A.1.39 B.2.39

Table 19-1, cont.				
Item No.	**Commitment**	**Implementation Schedule**	**Source**	**Related LRA Section No./ Comments**
	• 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; • 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; • Include a requirement in program procedures to perform specific measurements and/or characterizations of structural deficiencies, based on the results of previous inspections and guidance from ACI 349.3R-96, Section 5.1.1, and ACI 201.1 68; • Include a requirement in program procedures to document in the program inspection report a comparison of the results of the program inspections with the results of the previous program inspection; • Include a requirement in program procedures to file the Structures Monitoring Program inspection reports in the BVPS document control system so that inspection results can be more effectively monitored;			

Table 19-1, cont.				
Item No.	**Commitment**	**Implementation Schedule**	**Source**	**Related LRA Section No./ Comments**
	• Include a requirement in program procedures to apply inspection acceptance criteria based on the results of past inspections and guidance from ACI 349.3R-96, Section 5.1.1, and ACI 201.1-68; and, • Include a requirement in program procedures that noted deficiencies will be reported using the Corrective Action Program.			
23	With the exception of flexible connections in ventilations systems, prior to the period of extended operation, FENOC will perform repetitive maintenance tasks to replace mechanical system elastomeric components that would otherwise be subject to aging management review. Subsequent frequencies of the repetitive replacements will be based on manufacturer recommendations and applicable operating experience.	May 27, 2027	FENOC Letter L-08-212	None
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
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

Table 19-1, cont.				
Item No.	**Commitment**	**Implementation Schedule**	**Source**	**Related LRA Section No./ Comments**
26	Enhance the Metal Fatigue of the Reactor Coolant Pressure Boundary Program to: • Add a requirement that fatigue will be managed for the NUREG/CR-6260 locations. This requirement will provide that management is accomplished by one or more of the following: 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. • Add a requirement that provides for reanalysis, repair, or replacement of the Unit 2 steam generator secondary manway bolts and the steam generator tubes such that the design bases of these components are not exceeded for the period of extended operation. • Add a requirement to monitor Unit 2 transients where the 60 year projected cycles are used in the environmental fatigue evaluations, and establish an administration limit that is equal to or less than the 60-year projected cycles number.	May 27, 2027	LRA and FENOC Letter L-08-209	B.2.27

Table 19-1, cont.				
Item No.	**Commitment**	**Implementation Schedule**	**Source**	**Related LRA Section No./ Comments**
27	With the exception of underground GeoFlex® fuel oil piping, prior to the period of extended operation, FENOC will perform repetitive maintenance tasks to replace, or to test and replace on condition, mechanical system polymer components that would otherwise be subject to aging management review. Subsequent frequencies of the repetitive tests/replacements will be based on manufacturer recommendations and applicable operating experience.	May 27, 2027	FENOC Letter L-08-209, FENOC Letter L-08-212, and FENOC Letter L-08-376	None
28	Confirm the effectiveness of the new license renewal aging management programs based on the incorporation of operating experience by performing a program self assessment of all new license renewal aging management programs. [See NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," Appendix A, "Branch Technical Positions," Section A.1.2.3.10, Items 1 and 2.]	May 27, 2032	FENOC Letter L-08-209 and FENOC Letter L-08-226	B.2.8 B.2.10 B.2.11 B.2.12 B.2.13 B.2.15 B.2.21 B.2.22 B.2.26 B.2.30 B.2.31 B.2.36 B.2.41
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
30	As part of the Reactor Vessel Integrity Program, FENOC will store and maintain Unit 2 standby surveillance capsules in a condition that would permit their future use through the end of the period of extended operation.	Within 30 days following receipt of renewed license	FENOC Letter L-08-143	B.2.35

Table 19-1, cont.				
Item No.	**Commitment**	**Implementation Schedule**	**Source**	**Related LRA Section No./ Comments**
31	Enhance the Open-Cycle Cooling Water System Program to: • Assess the internal condition of buried piping by opportunistic inspections of header piping internals during removal of expansion joints and inline valves in the headers. Evaluation of inspection results will be documented and trended.	May 27, 2027	LRA and FENOC Letter L-08-262	A.1.32 B.2.32
32	Implement "needed actions" of MRP-146. These actions include screening, detailed analysis, inspections and temperature monitoring in accordance with the guidelines of MRP 146. FENOC has completed screening of the BVPS RCS branch lines.	FENOC will perform detailed evaluations (analysis, inspections and/or monitoring) in accordance with MRP-146 schedule requirements, or as established by the MRP committee.	FENOC Letter L-08-287	None
33	Supplemental volumetric examinations will be performed on the Unit 2 containment liner prior to the period of extended operation. A minimum of seventy-five (one foot square) randomly selected (as described in FENOC Letter L-09-205) sample locations will be examined (as described in FENOC Letter L-09-243). If degradation is identified, it will be addressed through the corrective action program (as described in FENOC Letter L-09-243).	Examinations will commence by the end of the Unit 2 Refueling Outage in 2011. The random sample plan will be completed by May 27, 2027.	FENOC Letter L-09-205, FENOC Letter L-09-243 and FENOC Letter L-09-244	None

Table 19-1, cont.				
Item No.	**Commitment**	**Implementation Schedule**	**Source**	**Related LRA Section No./ Comments**
34	Supplemental volumetric examinations will be performed on the Unit 2 containment liner. A minimum of 8 non-randomly selected locations will be examined, focusing on areas most likely to experience degradation based on past operating experience (as described in FENOC Letter L-09-242). If degradation is identified, it will be addressed through the corrective action program.	May 27, 2027	FENOC Letter L-09-*205,* FENOC Letter L-09-242 and FENOC Letter L-09-245	None
35	A summary of results for each phase of volumetric testing (described in Unit 2 Commitments No. 33 and No. 34) will be documented in a letter to the NRC.	May 27, 2027	FENOC Letter L-09-242 and FENOC Letter L-09-245	None
36	FENOC will evaluate if an appropriate/applicable statistical method exists to gain additional insight into potential liner degradation. Data gathered will be evaluated and used to determine the general state of the liner.	May 27, 2027	FENOC Letter L-09-242 and FENOC Letter L-09-243	None