SECTION 16

MANAGING THE EFFECTS OF COMPONENT AGING

TABLE OF CONTENTS

TABLE OF CONTENTS (CONT'D)

LIST OF TABLES

SECTION 16

MANAGING THE EFFECTS OF COMPONENT AGING

16.0 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. 1 that was scheduled to expire on January 29, 2016.

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 January 29, 2036.

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 16-1, Unit 1 License Renewal Commitments.

16.1 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.

16.1.1 <u>10 CFR Part 50, Appendix J Program</u>

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 16.1-2], with conditions and limitations specified in NEI 94-01, Revision 2-A.

16.1.2 <u>ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program</u>

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 16.1.42) where applicable.

16.1.3 <u>ASME Section XI, Subsection IWE Program</u>

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.

16.1.4 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 16.1-5].

16.1.5 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.

16.1.6 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 16.1-6], and EPRI NP-5769, *Degradation and Failure of Bolting in Nuclear Power Plants* [Reference 16.1-7]. Also, it implements industry recommendations for comprehensive bolting maintenance, as delineated in EPRI TR-104213, *Bolted Joint Maintenance & Application Guide* [Reference 16.1-8], 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

16.1.7 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 16.1-9].

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

16.1.8 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.

16.1.9 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, and Emergency Response Facility diesel generator cooling water).

These systems are closed cooling loops with controlled chemistry, consistent with the NUREG-1801 [Reference 16.1-33] 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.

16.1.10 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 16.1.14), 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.

16.1.11 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.

16.1.12 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.

16.1.13 Not Applicable (Unit 2 only)

16.1.14 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 16.1-33], 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* [Reference 16.1-34], no additional information is required to address GSI 168, *Environmental Qualification of Low-Voltage Instrumentation and Control (I&C) Cables* [Reference 16.1-35].

16.1.15 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.

16.1.16 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.

16.1.17 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 16.1-11]. 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.

16.1.18 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 16.1-12]. 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.

16.1.19 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 16.1-13].

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.

16.1.20 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 16.1.30) will be used to verify the effectiveness of the Fuel Oil Chemistry Program.

16.1.21 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.

16.1.22 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.

16.1.23 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 16.1-14].

16.1.24 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 16.1.30) will be used to verify the effectiveness of the Lubricating Oil Analysis Program.

16.1.25 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 16.1-15], and NRC Information Notice 87-67, *Lessons Learned from Regional Inspections of Licensee Actions in Response to IE Bulletin 80-11* [Reference 16.1-16]. 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 16.1.39) 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.

16.1.26 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. The Unit 2 480-VAC metal enclosed bus bolted connections are checked 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.

16.1.27 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* [Ref. 16.1-17], and the guidance of EPRI Technical Report MRP-47, *Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application* [Ref. 16.1-18]. 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* [Ref. 16.1-19], and in NUREG/CR 5704, *Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels* [Ref. 16.1-20].

Aging Management Program Elements

The results of an evaluation of each of the 10 aging management program elements described in NUREG-1801 [Reference 16.1-33], 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. Also, included is BVPS Unit 1 Surge Line.

  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, reactor trip from full

power, inadvertent auxiliary spray, and RCS cold overpressurization. Supplemental transients were also identified by the program. These supplemental transients include pressurizer insurge transient, selected Chemical and Volume Control System transients, safety injection actuation, Auxiliary Feedwater injections and RHR system actuation. Of the supplemental transients, only the pressurizer insurge and safety injection actuation transients require monitoring.

The number of critical transient occurrences is periodically reviewed (fuel cycle 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 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.

  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.

16.1.28 [Deleted]

16.1.29 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 16.1-21], 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 16.1-22]. Detection of cracking is accomplished through implementation of a combination of bare metal visual examination (external surface of head) and non-visual examination techniques.

16.1.30 One-Time Inspection Program

The One-Time Inspection Program requires one-time inspections to verify effectiveness of the Water Chemistry Program (Section 16.1.42), the Fuel Oil Chemistry Program (Section 16.1.20), and the Lubricating Oil Analysis Program (Section 16.1.24). 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 selected bottoms of tanks that sit on concrete pads (by volumetric examination); and,
- Cracking of aluminum alloy moisture separators associated with the Unit 1 Emergency Diesel Generator Air Start System.

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.

16.1.31 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 all 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.

16.1.32 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 16.1-23], including Supplement 1. This program manages the aging effects on the open-cycle cooling water systems such that the systems will be able to fulfill their intended function during the period of extended operation. The program includes surveillance and control techniques to manage aging effects caused by biofouling, corrosion, erosion, protective coating failures, and silting in the River Water System or structures and components serviced by the system.

16.1.33 PWR Vessel Internals Program

The PWR Vessel Internals aging management program relies on implementation of the inspection and evaluation guidelines in EPRI Technical Report MRP-227, “Pressurized Water Reactor Internals Inspection and Evaluation Guideline” and EPRI Technical Report MRP-228, “Inspection Standard for Pressurized Water Reactor Internals” to manage the aging effects on the reactor vessel internal components. This program is used to manage: (a) cracking, including stress corrosion cracking (SCC), primary water stress corrosion cracking (PWSCC), irradiation assisted stress corrosion cracking (IASCC), and cracking due to fatigue/cyclical loading; (b) loss of material induced by wear; (c) loss of fracture toughness due to either thermal aging, neutron irradiation embrittlement, or void swelling; (d) dimensional changes due to void swelling or distortion; and (e) loss of preload due to thermal and irradiation-enhanced stress relaxation or creep.

Beaver Valley Power Station participates in the industry programs for investigating and managing aging effects in reactor internals. The recommended activities provided in MRP-227 and additional plant-specific activities are implemented in accordance with Nuclear Energy Institute (NEI) 03-08, “Guidelines for the Management of Materials Issues.”

16.1.34 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 16.1-24].

16.1.35 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 16.1-25] (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 (RTPTS) 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.

16.1.36 Selective Leaching of Materials Inspection Program

The Selective leaching program identifies loss of material and degraded component integrity. Selective leaching is a galvanic corrosion mechanism that removes one of the alloying elements from an alloy material over time, causing a significant reduction in material strength. The purpose of this condition monitoring program is to demonstrate the absence or presence of selective leaching before a loss of component intended function. Selected components susceptible to selective leaching include components and commodities (such as piping, pump casings, valve bodies and heat exchanger components) made of gray cast iron and copper alloys (except for inhibited brass) that contain greater than 15 percent zinc (> 15% Zn) or greater than 8 percent aluminum (>8% Al in the case of aluminum-bronze) exposed to a raw water, closed cooling water, treated water, or ground water environment.

This program consists of a one-time inspection of sample populations from the components described above prior to the period of extended operation. The corrective action process is used for the unacceptable inspection findings. The resolution evaluates expansion of the inspection sample size, locations, and frequency in augmented inspections based on the one-time inspection and operating experience. Inspections may be conducted by visual inspection, qualitative hardness test (such as scraping or chipping of the susceptible surface) laboratory microscopic examinations or other industry accepted methodologies. Program inspection reports are kept in retrievable form.

Since selective leaching was identified prior to the period of extended operation in the majority of buried gray cast iron fire protection system, it is assumed to occur throughout the buried gray cast iron fire protection system. Therefore, condition monitoring for this set of components will be captured within the fire protection and fire water supply programs in lieu of the selective leaching program.

16.1.37 Not Applicable (Unit 2 only)

16.1.38 Steam Generator Tube Integrity Program

The Steam Generator Tube Integrity Program is based on NEI 97-06, *Steam Generator Program Guidelines* [Reference 16.1-27]. 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.

16.1.39 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 16.1-28] and Regulatory Guide 1.160, *Monitoring the Effectiveness of Maintenance at Nuclear Power Plants* [Reference 16.1-29].

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.

16.1.40 [Deleted]

16.1.41 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 16.1-30], are adequate for all pump casings and valve bodies.

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

16.1.42 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 16.1-33] that require augmentation of the Water Chemistry Program.

16.1.43 Boral Surveillance Program

The Boral Surveillance Program is an existing plant-specific condition monitoring program for which there is no comparable NUREG-1801 [Reference 16.1-33] aging management program. The program manages the neutron absorbing function of the BVPS Unit 1 High Density Spent Fuel Storage Racks by the removal and testing of sample Boral neutron absorber coupons. Coupon analysis is performed by a vendor, and recommendations based on the analysis are provided to the company.

The purpose of the program is to characterize certain properties of the Boral in the storage racks to assure its capability to fulfill its intended function, and to assure that assumptions made in the Fuel Pool criticality analysis remain valid. Because the test coupons are located and configured to ensure exposure to higher-than-average levels of gamma radiation, data gathered by the program represent accelerated use, and there is reasonable assurance that degradation will be detected and corrective actions taken prior to a loss of intended function.

16.1.44 References

16.1-2 NEI 94-01, *Industry Guidance for Implementing Performance-Based Options of 10 CFR Part 50 Appendix J,* Rev. 3-A.

16.1-3 [Deleted]

16.1-4 [Deleted]

16.1-5 *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.

16.1-6 NUREG-1339, *Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants,* October 17, 1991.

16.1-7 EPRI NP-5769, *Degradation and Failure of Bolting in Nuclear Power Plants,* May 5, 1988.

16.1-8 EPRI TR-104213, *Bolted Joint Maintenance & Application Guide,* December 1, 1995.

16.1-9 NRC Generic Letter 88-05, *Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants,* March 17, 1988.

16.1-10 [Deleted]

16.1-11 National Fire Protection Association NFPA 25, *Standard for the Inspection, Testing and Maintenance of Water-Based Fire Protection Systems,* 2002 Edition.

16.1-12 NSAC-202L-R2, *Recommendations for an Effective Flow Accelerated Corrosion Program,* April 1999.

16.1-13 NRC IE Bulletin 88-09, *Thimble Tube Thinning in Westinghouse Reactors,* July 26, 1988.

16.1-14 NUREG-0612, *Control of Heavy Loads at Nuclear Power Plants,* July 1980.

16.1-15 NRC IE Bulletin 80-11, *Masonry Wall Design,* May 8, 1980.

16.1-16 NRC Information Notice 87-67, *Lessons Learned from Regional Inspections of Licensee Actions in Response to IE Bulletin 80-11,* December 31, 1987.

16.1-17 NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves for Selected Nuclear Power Plant Components,* February 28, 1995.

16.1-18 EPRI Technical Report MRP-47, *Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application,* September 1, 2005.

References to Section 16.1 (Continued)

16.1-19 NUREG/CR-6583, *Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low Alloy Steels,* February 1998.

16.1-20 NUREG/CR-5704, *Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels,* April 1999.

16.1-21 NRC Order EA 03-009, *Issuance of Order Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors,* February 11, 2003.

16.1-22 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.

16.1-23 NRC Generic Letter 89-13, *Service Water System Problems Affecting Safety-Related Equipment,* including Supplement 1, July 18, 1989.

16.1-24 Regulatory Guide 1.65, *Materials and Inspections for Reactor Vessel Closure Studs,* October 1973.

16.1-25 ASTM Standard E 185-82, *Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels,* June 2002.

16.1-26 [Deleted]

16.1-27 NEI 97-06, *Steam Generator Program Guidelines,* Rev. 2, May 2005.

16.1-28 NUMARC 93-01, *Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,* Rev. 3, October 8, 1999.

16.1-29 Regulatory Guide 1.160, *Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,* Rev. 2, March 1997.

16.1-30 ASME Code Case N-481, *Alternate Examination Requirements for Cast Austenitic Pump Casings,* May 20, 1998.

16.1-31 [Deleted]

16.1-32 [Deleted]

16.1-33 NUREG-1801, *Generic Aging Lessons Learned (GALL) Report*, Rev. 1, September 2005.

16.1-34 RIS 2003-09, *Environmental Qualification of Low-Voltage Instrumentation and Control Cables*, May 2, 2003.

16.1-35 GSI 168, *Environmental Qualification of Low-Voltage Instrumentation and Control (I&C) Cables*, June 30, 2004.

16.2 EVALUATION SUMMARIES OF UNIT 1 TIME-LIMITED AGING ANALYSES

16.2.1 Introduction

Time-limited aging analyses (TLAAs) are defined in 10 CFR 54.3 [Reference 16.2-3] 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 16.2.2)
- Metal Fatigue (Section 16.2.3)
- Environmental Qualification (EQ) of Electric Equipment (Section 16.2.4)
- Containment Liner Plate, Metal Containment, and Penetrations Fatigue (Section 16.2.5)
- Other Plant-Specific Time-Limited Aging Analyses (Section 16.2.6)
- References (Section 16.2.7)

16.2.2 Reactor Vessel Neutron Embrittlement

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

- Neutron Fluence Values (Section 16.2.2.1)
- Pressurized Thermal Shock (Section 16.2.2.2)
- Charpy Upper Shelf Energy (Section 16.2.2.3)
- Pressure-Temperature Limits (Section 16.2.2.4)

16.2.2.1 Neutron Fluence Values

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

In the Fall of 2013, Surveillance Capsule X was pulled and the analysis was documented in WCAP-17896, *Analysis of Capsule X from the First Energy Nuclear Operating Company Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program* [Reference 16.2-33]. For the 60-year license, WCAP-18102 Rev 1, *Beaver Valley Unit 1 Heatup and Cooldown Limit Curve for Normal Operation* [Reference 16.2-34], 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 Regulatory Guide 1.190, *Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence* [Reference 16.2-6].

The fluence projections include fuel cycle-specific calculated neutron exposures through the end of Cycle 24 (Fall 2016), as well as future projections 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 18. Neutron exposure projections beyond the end of Cycle 24 were based on the core source distributions and associated plant characteristics of Cycles 22 through 24 in conjunction with the uprated power level.

16.2.2.2 Pressurized Thermal Shock

In the Fall of 2013, Surveillance Capsule X was pulled and the analysis was documented in WCAP-17896 [Reference 16.2-33]. For the 60-year license, WCAP-18102 Rev 1, App E [Reference 16.2-34] documents the EOLE analysis for pressurized thermal shock (PTS).

Using the prescribed PTS Rule (10 CFR 50.61 [Reference 16.2-7]) methodology, reference temperature for pressurized thermal shock (RT_{PTS}) values were generated for beltline and extended beltline region materials of the BVPS Unit 1 Reactor Vessel for fluence values at EOLE (50 EFPY). The projected RT_{PTS} values for EOLE (50 EFPY) meet the 10 CFR 50.61 screening criteria for beltline and extended beltline materials.

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

16.2.2.3 Charpy Upper Shelf Energy

In the fall of 2013, Surveillance Capsule X was pulled and the analysis was documented in WCAP-17896 [Reference 16.2-33]. For the 60-year license, WCAP-18102 [Reference 16.2-34] and Westinghouse letter LTR-SDA-17-017 Rev 0 [Reference 16.2-35] document the EOLE analysis for Charpy upper-shelf energy (C_VUSE).

For Unit 1, there exists material surveillance data for Reactor Vessel lower shell plate B6903-1 (heat C6317-1) and the intermediate shell longitudinal weld (heat 305424). The measured drops in C_VUSE for each of these material heats was plotted on Figure 2 of Regulatory Guide 1.99, *Radiation Embrittlement of Reactor Vessel Materials* [Reference 16.2-8], 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 50 EFPY. Therefore, the Unit 1 C_VUSE analysis has been dispositioned in accordance with 10 CFR 54.21(c)(1)(ii).

16.2.2.4 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 16.2-7], require BVPS to operate within the currently licensed P-T limit curves. These curves are required to be maintained and updated as necessary to maintain plant operation consistent with 10 CFR 50. The Reactor Vessel Integrity Program will maintain the P-T limit curves for Unit 1 for the period of extended operation. Therefore, the Unit 1 P-T limit curves TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

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

16.2.3 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 16.2.3.1)
- Non-Class 1 Fatigue Evaluations (Section 16.2.3.2)
- Generic Industry Issues on Fatigue (Section 16.2.3.3)

16.2.3.1 Class 1 Fatigue Evaluations

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

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

16.2.3.1.1 Unit 1 Pressurizer

In 1999, the analysis of the Unit 1 pressurizer, lower shell and related components was revised to address improvements to the insurge/outsurge transients identified by the Westinghouse Owners Group. Plant operating procedures were revised to follow the guidance of the Westinghouse Owners Group and to minimize the impact of potential insurges. Prior to the 1999 reanalysis, BVPS Unit 1 had experienced several pressurizer spray transients that challenged the analytical and Technical Specification limit of 320°F difference between the spray line temperature and the pressurizer steam space temperature. Revised transients for initial spray flow were incorporated into the analysis. In 2005, BVPS decided to further revise the operating procedures to optimize the plant shutdown and startup processes. The BVPS optimized procedures have been shown to meet all recommendations of the Westinghouse Owners Group and have virtually eliminated the potential for insurges. Next, the Extended

Power Uprate Project evaluated the revised Uprate transients against the previous analysis. The cumulative usage factors associated with the Unit 1 pressurizer are less than 1.0. Since the 60-year projected operational cycles were used in determining that the pressurizer design fatigue analysis remains valid for 60 years, the Metal Fatigue of Reactor Coolant Pressure Boundary Program must continue to be used to validate the assumptions used in the evaluation. In addition, the pressurizer insurge cycle assumptions used in the pressurizer analysis require validation for the period of extended operation. The Metal Fatigue of Reactor Coolant Pressure Boundary Program identifies the pressurizer insurge transient as a supplemental transient that requires monitoring. Therefore, the pressurizer fatigue TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(iii).

16.2.3.2 Non-Class 1 Fatigue Evaluations

16.2.3.2.1 Piping and In-Line Components

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

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

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

16.2.3.2.2 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 (i.e., 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 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 16.2.3.2.1), 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. The company 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).

16.2.3.3 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.

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

NRC Bulletin 88-11, *Pressurizer Surge Line Thermal Stratification* [Reference 16.2-9], required a plant-specific or generic analysis demonstrating that the pressurizer surge line meets the applicable design code requirements considering the effects of thermal stratification.

In response to the Bulletin, BVPS submitted a plant-specific analysis, WCAP-12727, *Evaluation of Thermal Stratification for the Beaver Valley Unit 1 Pressurizer Surge Line* [Reference 16.2-10], to the NRC. The NRC approved [Reference 16.2-11] this evaluation. WCAP-12727 determined the effects of thermal stratification in the surge line through the imposition of defined thermal stratification cycles upon the stress and fatigue evaluations. The stratification cycles incorporated into the cumulative usage factor determination are defined by the 200 heatup and cooldown design transients. Therefore, this NRC Bulletin 88-11 analysis is a TLAA in accordance with 10 CFR 54.3.

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

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

16.2.3.3.2 Effects of Primary Coolant Environment on Fatigue Life

Test data indicate that certain environmental conditions (such as temperature, oxygen content, and strain rate) in the primary systems of light water reactors could result in greater susceptibility to fatigue than would be predicted by fatigue analyses based on the ASME Section III design fatigue curves. One NRC study, documented in NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components* [Reference 16.2-12], applied the fatigue design curves that incorporated environmental effects to several plant designs. The results of studies performed on this topic, including NUREG/CR-6260, were summarized in Generic Safety Issue (GSI)-190, *Fatigue Evaluation of Metal Components for 60-Year Plant Life* [Reference 16.2-13]. In closing GSI-190, regarding the effects of a reactor water environment on fatigue life, the NRC concluded that licensees should address the effects of the coolant environment on component fatigue life as aging management programs are formulated in support of license renewal.

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

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

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

The NUREG/CR-6260 locations were evaluated using the guidance of NUREG/CR-6583, *Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low Alloy Steels* [Reference 16.2-14], and NUREG/CR-5704, *Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels* [Reference 16.2-15]. These reports describe the use of a fatigue life correction factor (F_{en}) to express the effects of the reactor coolant environment upon the material fatigue life. The expression for F_{en} was determined through experimental and statistical data. F_{en} for carbon and low alloy steel is a function of fluid service temperature, material sulfur content, fluid dissolved oxygen, and strain rate. For austenitic stainless steel, F_{en} is a function of fluid service temperature, fluid dissolved oxygen, and strain rate. The cumulative usage factor which includes environmental effects (U_{env}) is determined from the existing 60-year cumulative usage factor (U_{60}) through the use of the fatigue life correction factor:

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

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

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

As discussed in Section 16.2.3.1, 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).

16.2.4 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).

16.2.5 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 16.2.5.1)
- Containment Liner Corrosion Allowance (Section 16.2.5.2)
- Containment Liner Penetration Fatigue (Section 16.2.5.3)

16.2.5.1 Containment Liner Fatigue

The Unit 1 containment liner stress analysis determines a fatigue usage factor based on specific design cyclic loads in accordance with paragraph N-415.2 of the 1968 Edition of ASME Section III. These design loads include 1000 cycles of pressure variation due to normal operations (startup and shutdown), 4000 cycles of temperature variation due to normal operations (startup and shutdown), and 20 cycles of design basis earthquake (DBE). The fatigue analysis determined the stress due to the combination of thermal, normal operating and DBE loadings. That combination was then considered as 4000 cycles of a fluctuation from the operating condition (including DBE) to the zero stress state in determining the cumulative usage factor (CUF). The CUF was determined to be significantly less than 1.0.

The anticipated occurrences of these cycles are described in Table 5.2-13 of the Unit 1 UFSAR as follows:

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

As shown above the 60-year anticipated occurrences of pressure cycles, temperature cycles and OBE cycles are bounded by the 4000 analyzed cycles. Therefore, the Unit 1 Containment liner fatigue TLAA has been dispositioned in accordance with 10 CFR 54.21(c)(1)(i).

16.2.5.2 Containment Liner Corrosion Allowance

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

During a Unit 1 shutdown in 1991, it was determined that 27 vent plugs in the Containment floor liner test channels were missing. The missing test channel vent plugs allowed moisture and condensation inside the test channels, leading to minor corrosion of the liner. BVPS evaluated the test channels to determine the impact to the Containment liner, and submitted the results of the evaluations to the NRC as Amendments 165 and 47, Unit 1 and Unit 2 respectively, to the operating licenses. These amendments were approved by the NRC and documented in an SER [Reference 16.2-16]. After further evaluation, it was concluded that these initial evaluations contained some nonconservative assumptions with regard to the corrosion rates in the test channels. BVPS took corrective action to arrest the corrosion rate in the affected test channels, including inerting and sealing the test channels. The further evaluation and corrective actions are documented in a 1992 Letter to the NRC [Reference 16.2-17]. These corrosion rate analyses meet the 10 CFR 54.3 requirements as TLAAs and must be evaluated for the period of extended operation.

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

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

16.2.5.3 Containment Liner Penetration Fatigue

16.2.5.3.1 Equipment Hatch

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

16.2.5.3.2 Fuel Transfer Tube

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

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

16.2.5.3.3 Containment Penetration Bellows

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

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

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

16.2.6 Other Plant-Specific Time-Limited Aging Analyses

The plant-specific TLAAs summarized in this section include:

- Piping Subsurface Indications (Section 16.2.6.1)
- Reactor Vessel Underclad Cracking (Section 16.2.6.2)
- Leak Before Break (Section 16.2.6.3)
- Crane Load Cycles (Section 16.2.6.4)

16.2.6.1 Piping Subsurface Indications

During a Unit 1 inservice inspection performed in the Cycle 11 Refueling Outage (March - May, 1996), an indication was identified on the RCS loop C cold leg between an elbow and a section of straight pipe which exceeded the ASME Code, Section XI, subsection IWB-3500 acceptance criteria. This section of pipe is Class 1 cast austenitic stainless steel (CASS) piping. Subsequently, an analysis was performed to ensure that this indication would remain within ASME Code, Section XI, Appendix C evaluation acceptance standards. This evaluation, approved by the NRC [Reference 16.2-18], concluded that the postulated flaw met the applicable requirements with significant margins of safety to the end of the service lifetime. This flaw growth evaluation is a TLAA because it contained two parameters that are based on the service life of the piping, namely thermal aging and fatigue transient cycles.

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

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

16.2.6.2 Reactor Vessel Underclad Cracking

WCAP-15338-A, *A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants* [Reference 16.2-20], evaluates the impact of cracks beneath austenitic stainless steel weld cladding on reactor pressure vessel integrity for 60 years of operation.

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

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

16.2.6.3 Leak Before Break

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

For the LBB analyses discussed in the following two subsections, the only consideration that could be influenced by time is the accumulation of actual fatigue transient cycles. The cycle assumptions used in the analyses are conservative compared to the BVPS original design transients. The BVPS original design basis transients including design cycles are identified in Table 4.1-10 of the UFSAR. BVPS has reviewed the design cycles against 60-year projected operational cycles and has determined that the design cycles are bounding for the period of

extended operation. Since the 60-year projected operational cycles were used in determining that the flaw growth analyses remain valid for 60 years, the Metal Fatigue of Reactor Coolant Pressure Boundary Program must continue to be used to validate the assumptions used in the evaluations. Therefore, the Unit 1 flaw growth TLAAs have been dispositioned in accordance with 10 CFR 54.21(c)(1)(i) and 10 CFR 54.21(c)(1)(iii).

16.2.6.3.1 Main Coolant Loop Piping Leak Before Break

The original LBB evaluation for the main coolant loop piping was documented in WCAP-11317, *Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Beaver Valley Unit 1* [Reference 16.2-21]. This evaluation (including Supplements 1 and 2) was approved by the NRC in a Safety Evaluation Report [Reference 16.2-22] in 1987.

Supplement 3 to WCAP-11317 [Reference 16.2-29] was issued in 2005 to incorporate the Power Uprate, Steam Generator Replacement and License Renewal projects.

The reactor coolant loop piping was reanalyzed in 2011 for LBB in WCAP-17408-P [Reference 16.2-30], to incorporate the latest piping loads, operating conditions and Standard Review Plan criteria. WCAP-17408-P supersedes all previous LBB analyses for the main coolant loop piping.

16.2.6.3.2 Pressurizer Surge Line Piping Leak Before Break

The LBB evaluation for the pressurizer surge line piping was documented in WCAP-12727, *Evaluation of Thermal Stratification for the Beaver Valley Unit 1 Pressurizer Surge Line* [Reference 16.2-23]. This evaluation was approved by the NRC in a Safety Evaluation Report [Reference 16.2-24] in 1991.

Supplements 1 and 2 to WCAP-12727 [Reference 16.2-31] were issued in 2005 to incorporate the Power Uprate and Steam Generator Replacement projects.

The pressurizer surge line piping was reanalyzed in 2011 for LBB in WCAP-17406-P [Reference 16.2-32], to incorporate the latest piping loads, operating conditions and Standard Review Plan criteria. WCAP-17406-P supersedes all previous LBB analyses for the pressurizer surge line piping. WCAP-12727 remains valid for analyses with respect to surge line stratification.

16.2.6.4 Crane Load Cycles

In the response to NUREG-0612, *Control of Heavy Loads at Nuclear Power Plants* [Reference 16.2-26], BVPS determined that two cranes, the single failure proof fuel cask crane (CR-15), and the moveable platform and hoists (CR-27), were designed to comply with Crane Manufacturers Association of America Specification #70 (CMAA-70). CR-27 was designed to Reference 16.2-27 and CR-15 was designed to Reference 16.2-25. Therefore, these cranes have a TLAA associated with their design calculations. The Single-Failure-Proof fuel cask crane (CR-15) is also in compliance with NUREG-0554, Single-Failure-Proof Cranes for Nuclear Power Plants [Reference 16.2-19].

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).

16.2.7 References

16.2-1 BVPS License Renewal Application, FENOC Letter L-07-113, August 27, 2007.

16.2-2 NRC SER for BVPS License Renewal, Volumes 1 & 2, with Supplement, ML093020276, ML093000278, and ML092570014.

16.2-3 10 CFR 54, *Requirements for Renewal of Operating Licenses for Nuclear Power Plants.*

16.2-4 WCAP-15571, *Analysis of Capsule Y from First Energy Company Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program,* Rev. 0.

16.2-5 WCAP-15571 Supplement 1, Revision 2, *Analysis of Capsule Y from Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program*, September 2011.

16.2-6 Regulatory Guide 1.190, *Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,* March 2001.

16.2-7 10 CFR 50, *Domestic Licensing of Production and Utilization Facilities.*

16.2-8 Regulatory Guide 1.99, *Radiation Embrittlement of Reactor Vessel Materials*, Rev. 2.

16.2-9 NRC Bulletin 88-11, *Pressurizer Surge Line Thermal Stratification*, December 20, 1988.

16.2-10 WCAP-12727, *Evaluation of Thermal Stratification for the Beaver Valley Unit 1 Pressurizer Surge Line*, Rev. 0.

16.2-11 De Agazio, Albert W. (NRC), Letter to John D. Sieber (BVPS), *Approval of Leak-Before-Break Analysis* (TAC No. 72110), May 2, 1991.

16.2-12 NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components*, February 1995.

16.2-13 Generic Safety Issue (GSI)-190, *Fatigue Evaluation of Metal Components for 60-Year Plant Life*, Rev. 2.

16.2-14 NUREG/CR-6583, *Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low Alloy Steels*, February 1998.

16.2-15 NUREG/CR-5704, *Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels*, March 1999.

16.2-16 De Agazio, Albert W. (NRC), Letter to J. D. Sieber (BVPS), Beaver Valley Units 1 and 2 - *Issuance of Amendments 165 and 47: Containment Structural Integrity - Change Request Nos. 181/45*, June 23, 1992.

References to Section 16.2 (Continued)

16.2-17 Sieber, J. D. (BVPS), Letter to NRC, *Beaver Valley Power Station, Unit No. 1 and No. 2, BV-1 Docket No. 50-334, License No. DPR-66, BV-2 Docket No. 50-412, License No. NPF-73, Revision to SER for Amendments 165 and 47*, December 30, 1992.

16.2-18 Brinkman, Donald S. (NRC), Letter to J.E. Cross (BVPS), *Evaluation of Flaw Indication in Reactor Coolant System (RCS) Cold Leg Pipe Weld, Beaver Valley Power Station, Unit No. 1 (BVPS-1)*, May 1, 1996.

16.2-19 NUREG-0554, *Single-Failure-Proof Cranes for Nuclear Power Plants*, May 1979.

16.2-20 WCAP-15338-A, *A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants*, October 2002.

16.2-21 WCAP-11317, *Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Beaver Valley Unit 1*, March 1987 (including Supplements 1 and 2).

16.2-22 Tam, Peter S. (NRC), Letter to J.D. Sieber (BVPS), *Beaver Valley Unit 1 - Removal of Large-Bore Snubbers from Primary Coolant Loops*, December 9, 1987.

16.2-23 WCAP-12727, *Evaluation of Thermal Stratification for the Beaver Valley Unit 1 Pressurizer Surge Line*, November 1990.

16.2-24 De Agazio, Albert W. (NRC), Letter to J. D. Sieber (BVPS), *Approval of Leak-Before-Break Analysis*, May 2, 1991.

16.2-25 Crane Manufacturers Association of America Specification #70 (CMAA-70-210) *Specifications for Top Running Bridge and Gantry Type Multiple Girder Electric Overhead Traveling Cranes*.

16.2-26 NUREG-0612, *Control of Heavy Loads at Nuclear Power Plants*, July 1980.

16.2-27 Crane Manufacturers Association of America Specification #70 (CMAA-70), *Specifications for Electric Overhead Traveling Cranes*, Revised 1983.

16.2-28 NUREG/CR-6934, *Fatigue Crack Flaw Tolerance in Nuclear Power Plant Piping - A Basis for Improvements to ASME Code Section XI Appendix L*, May 2007.

16.2-29 WCAP-11317, Supplement 3, *EPU/RSG Project Update of Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Beaver Valley Unit 1,* January 2005.

16.2-30 WCAP-17408-P, *Leak-Before-Break Analysis Update for the Beaver Valley Unit 1 Primary Loop Piping*, April 2012.

References to Section 16.2 (Continued)

16.2-31 WCAP-12727, Supplement 1, *EPU/RSG Project Update of Evaluation of Thermal Stratification for the Beaver Valley Unit 1 Pressurizer Surge Line and Supplement 2 EPU/RSG Project Update of Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for Beaver Valley Unit 1*, both January 2005.

16.2-32 WCAP-17406-P, *Leak-Before-Break Analysis Update for the Beaver Valley Unit 1 Pressurizer Surge Line*, March 2012.

16.2-33 WCAP-17896-NP, *Analysis of Capsule X from the FirstEnergy Nuclear Operating Company Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program*, September 2014.

16.2-34 WCAP-18102-NP Revision 1, *Beaver Valley Unit 1 Heatup and Cooldown Limit Curves for Normal Operation*, February 2018.

16.2-35 LTR-SDA-17-017, *Beaver Valley Unit 1 Upper-Shelf Energy Values*, November 2014.

BVPS UFSAR UNIT 1

TABLES FOR CHAPTER 16

Table 16-1
UNIT 1 LICENSE RENEWAL COMMITMENTS

Table 16-1 identifies those actions committed to for BVPS Unit 1 in the BVPS License Renewal Application (LRA). These regulatory commitments will be tracked within the regulatory commitment management program.

Table 16-1				
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 January 29, 2016	LRA	A.1.8 B.2.8
2	Enhance the Closed-Cycle Cooling Water System Program to: • Add the diesel-driven fire pump (Unit 1 only) to the program; • Detail performance testing of heat exchangers and pumps, and provide direction to perform visual inspections of system components; • Identify closed-cycle cooling water system parameters that will be trended to determine if heat exchanger tube fouling or corrosion product buildup exists; • Control performance tests and perform visual inspections at the required frequency.	January 29, 2016	LRA	A.1.9 B.2.9

Table 16-1, 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 January 29, 2016	LRA and 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.	January 29, 2016	LRA	A.1.11 B.2.11
5	Implement the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program as described in LRA Section B.2.12.	January 29, 2016	LRA	A.1.12 B.2.12
6	Implement the External Surfaces Monitoring Program as described in LRA Section B.2.15.	January 29, 2016	LRA	A.1.15 B.2.15

Table 16-1, cont.				
Item No.	**Commitment**	**Implementation Schedule**	**Source**	**Related LRA Section No./ Comments**
7	Enhance the Fire Protection Program to: • Include a new attachment in the BVPS Fire Protection Program administrative procedure to address the Fire Protection Systems that are in scope for license renewal purposes; • Provide details of the NUREG-1801 inspection and testing guidelines, the plant implementation strategy, surveillance test and inspection frequencies *(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.	January 29, 2016	LRA and Letter L-08-375	A.1.16 B.2.16

Table 16-1, cont.				
Item No.	**Commitment**	**Implementation Schedule**	**Source**	**Related LRA Section No./ Comments**
8	Enhance the Fire Water System Program to: • Include a program requirement to perform flow test or inspection of all accessible fire water headers and piping during the period of extended operation at an interval determined by the Fire Protection System Engineer; • Include a program requirement that a representative number of fire water piping locations be identified if piping visual inspections are used as an alternative to non-intrusive testing; • Include a program requirement which allows test or inspection results from an accessible section of pipe to be extrapolated to an inaccessible, but similar section of pipe. If no similar section of accessible pipe is available, then alternative testing or inspection activities must be used; • Include a program requirement that, at least once prior to the period of extended operation, all accessible Fire Protection headers and piping shall be flow tested in accordance with NFPA 25 or visually/ultrasonically inspected; • 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 January 29, 2016	LRA	A.1.17 B.2.17

Table 16-1, cont.				
Item No.	**Commitment**	**Implementation Schedule**	**Source**	**Related LRA Section No./ Comments**
9	Enhance the Flux Thimble Tube Inspection Program to: • Include a requirement in the program procedure to state that, if a flux thimble tube cannot be inspected over the tube length (tube length that is subject to wear due to restriction or other defect), and cannot be shown by analysis to be satisfactory for continued service, the thimble tube must be removed from service to ensure the integrity of the Reactor Coolant System pressure boundary.	January 29, 2016	LRA	A.1.19 B.2.19
10	Enhance the Fuel Oil Chemistry Program to: • Revise the implementing procedure for sampling and testing the diesel-driven fire pump fuel oil storage tank (Unit 1 only) to include a test for particulate and accumulated water in addition to the test for sediment and water; • 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.	January 29, 2016	LRA, Letter L-08-262 and Letter L-08-316	A.1.20 B.2.20

Table 16-1, cont.				
Item No.	**Commitment**	**Implementation Schedule**	**Source**	**Related LRA Section No./ Comments**
11	Implement the Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program as described in LRA Section B.2.21. 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.	January 29, 2016	LRA; Letter L-08-262; Letter L-09-057; Letter L-09-138; and Letter L-09-151	A.1.21 B.2.21
12	Implement the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program as described in LRA Section B.2.22.	January 29, 2016	LRA	A.1.22 B.2.22

Table 16-1, cont.				
Item No.	**Commitment**	**Implementation Schedule**	**Source**	**Related LRA Section No./ Comments**
13	Enhance the Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems Program to: • Include guidance in the program administrative procedure to inspect for loss of material due to corrosion on Unit 1 crane and trolley structural components and rails; and, • Include guidance in the crane and hoist inspection procedures to inspect for loss of material due to corrosion on Unit 1 crane and trolley structural components and rails or extendable arms, as appropriate.	January 29, 2016	LRA	A.1.23 B.2.23
14	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.	January 29, 2016	LRA and Letter L-08-262	A.1.25 B.2.25
15	Regarding activities for managing the aging of nickel-alloy components and nickel-alloy clad components susceptible to primary water stress corrosion cracking - PWSCC (other than upper reactor vessel closure head nozzles and penetrations), BVPS commits to develop a plant-specific aging management program that will implement applicable: 1. NRC Orders, Bulletins and Generic Letters; and, 2. Staff-accepted industry guidelines.	January 29, 2016	LRA and Letter L-08-212	None

Table 16-1, cont.

Item No.	Commitment	Implementation Schedule	Source	Related LRA Section No./ Comments
16	Implement the One-Time Inspection Program as described in LRA Section B.2.30 and as amended by Change Notices 11-193 and 12-019.	Will be implemented within the 10 years prior to January 29, 2016	LRA	A.1.30 B.2.30
17	Implement the One-Time Inspection of ASME Code Class 1 Small-Bore Piping Program as described in LRA Section B.2.31.	Will be implemented within the 10 years prior to January 29, 2016	LRA	A.1.31 B.2.31
18	Regarding activities for managing the aging of Reactor Vessel internal components and structures, BVPS commits to: 1. Participate in the industry programs applicable to BVPS Unit 1 for investigating and managing aging effects on reactor internals; 2. Evaluate and implement the results of the industry programs as applicable to the BVPS Unit 1 reactor internals; and, 3. Upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for the BVPS Unit 1 reactor internals to the NRC for review and approval.	January 29, 2014	LRA and Letter L-08-212	None
19	Implement the Selective Leaching of Materials Program as described in LRA Section B.2.36 and as amended by Change Notice 11-248.	January 29, 2016	LRA	A.1.36 B.2.36

Table 16-1, cont.				
Item No.	**Commitment**	**Implementation Schedule**	**Source**	**Related LRA Section No./ Comments**
20	Enhance the Structures Monitoring Program to: • Include in program scope additional structures and structural components identified as having aging effects requiring management for license renewal; • Include inspection guidance in program implementing procedures to detect significant cracking in concrete surrounding the anchors of vibrating equipment; • 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;	January 29, 2016 for all enhancements except groundwater sampling (4th 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 and Letters L-08-181 and L-08-262	A.1.39 B.2.39

Table 16-1, cont.				
Item No.	**Commitment**	**Implementation Schedule**	**Source**	**Related LRA Section No./ Comments**
	• 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; • 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.			
21	With the exception of flexible connections in ventilations systems, prior to the period of extended operation, the company 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.	January 29, 2016	Letter L-08-212	None
22	Implement the Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program as described in LRA Section B.2.41.	January 29, 2016	LRA	A.1.41 B.2.41

Table 16-1, cont.				
Item No.	**Commitment**	**Implementation Schedule**	**Source**	**Related LRA Section No./ Comments**
23	Enhance the Water Chemistry Program to: • Change BVPS frequency for reactor coolant silica monitoring to once per week for Operational Modes 1 and 2, and once per day during heatup in Operational Modes 3 and 4 to be consistent with EPRI guidelines.	January 29, 2016	LRA	A.1.42 B.2.42
24	Prior to exceeding the PTS screening criteria for BVPS Unit 1, FENOC will select a flux reduction measure to manage PTS in accordance with the requirements of 10 CFR 50.61. A flux reduction plan will be submitted for NRC review and approval.	A flux reduction plan will be submitted at least 1 year prior to implementation of the flux reduction measure.	Letter L-08-124	A.2.2.2 4.2.2
25	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.	January 29, 2016	LRA Letter L-08-209	B.2.27

Table 16-1, cont.				
Item No.	**Commitment**	**Implementation Schedule**	**Source**	**Related LRA Section No./ Comments**
	• Add a requirement that establishes an administration limit of 600 cycles for the Unit 1 RHR system actuation transient. • Add a requirement to monitor Unit 1 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.			
26	Evaluate Unit 1 Extended Power Uprate operating experience prior to the period of extended operation for license renewal aging management program adjustments.	January 29, 2016	None	Appendix B.2
27	As part of the Reactor Vessel Integrity Program, FENOC will store and maintain Unit 1 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	Letter L-08-143	B.2.35
28	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.	January 29, 2016	Letter L-08-212	None

Table 16-1, cont.				
Item No.	**Commitment**	**Implementation Schedule**	**Source**	**Related LRA Section No./ Comments**
29	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.]	January 29, 2021	Letter L-08-226	B.2.8 B.2.10 B.2.11 B.2.12 B.2.15 B.2.21 B.2.22 B.2.30 B.2.31 B.2.36 B.2.41
30	Enhance the Open-Cycle Cooling Water System Program to: • Include in program scope the Post Accident Sample System heat exchanger (PAS-E-1) that is credited with a leakage boundary function; and, • 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.	January 29, 2016	LRA and Letter L-08-262	A.1.32 B.2.32
31	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.	Letter L-08-287	None

Table 16-1, cont.				
Item No.	**Commitment**	**Implementation Schedule**	**Source**	**Related LRA Section No./ Comments**
32	Supplemental volumetric examinations will be performed on the Unit 1 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).	Unit 1 inspections for the initial sample lot of a minimum of 75 random ultrasonic examinations will be completed in the next three refueling outages, beginning with the Unit 1 refueling outage in 2010. The random sample plan will be completed by January 29, 2016.	Letter L-09-205, Letter L-09-242 and Letter L-09-243	None
33	Supplemental volumetric examinations will be performed on the Unit 1 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.	Examinations will commence on-line, prior to the beginning of the Unit 1 Refueling Outage in 2010. Examinations will be completed by December 31, 2010.	Letter L-09-205, Letter L-09-242 and Letter L-09-245	None
34	A summary of results for each phase of volumetric testing (described in Unit 1 Commitments No. 32 and No. 33) will be documented in a letter to the NRC.	January 29, 2016	Letter L-09-242 and Letter L-09-245	None
35	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.	January 29, 2016	Letter L-09-242 and Letter L-09-243	None

Table 16-1, cont.				
Item No.	**Commitment**	**Implementation Schedule**	**Source**	**Related LRA Section No./ Comments**
36	Implement the Metal Enclosed Bus Program as described in LRA Section B.2.26, and as amended by UFSAR change notice CN 12-185.	January 29, 2016	LRA, CN 12-185	A.1.26 B.2.26