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18 LICENSE RENEWAL

18.0 AGING MANAGEMENT PROGRAMS AND TIME-LIMITED AGING ANALYSES ACTIVITIES

The integrated plant assessment for license renewal identified existing and new aging management programs necessary to provide reasonable assurance that components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis (CLB) for the period of extended operation. This chapter describes these programs and their planned implementation.

This chapter discusses the evaluation results for each of the plant-specific time-limited aging analyses (TLAAs) performed for license renewal. The evaluations have demonstrated 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 the effects of aging on the intended function(s) will be adequately managed for the period of extended operation. In addition, this chapter discusses the activities necessary to support the TLAAs.

No 10 CFR 50.12 exemptions involving a TLAA as defined in 10 CFR 54.3 were identified for Duane Arnold.

18.1 AGING MANAGEMENT PROGRAMS

This section provides summaries of the programs and activities, in alphabetical order, credited for managing the effects of aging. These aging management programs may not exist as discrete programs at Duane Arnold. In many cases, they exist as a compilation of various implementing documents that, when taken as a whole, satisfy the intent of NUREG-1800 and/or NUREG-1801 [References 18-2 and 18-3, respectively] attributes.

The Duane Arnold Quality Assurance Program implements the requirements of 10 CFR 50, Appendix B, and is consistent with the summary in Appendix A.2 of NUREG-1800. The elements of corrective action, confirmation process, and administrative controls in the Quality Assurance Program are applicable to both safety related and non-safety related systems, structures, and components that are subject to an aging management review.

18.1.1 10 CFR 50 APPENDIX J PROGRAM

The 10 CFR 50 Appendix J Program is a performance based containment leak rate test program. The program implements the guidelines contained in 10 CFR Appendix J, Option B. The program performs periodic inspections and surveillance testing of primary containment systems and components penetrating the primary containment to ensure that allowable leakage rates do not exceed Technical Specification requirements.

2013-015 This program is consistent with the ten elements of NUREG-1801 Rev. 1 XI.S4 and takes no exception to NUREG-1801 Rev. 1 XI.S4.

18.1.2 ABOVEGROUND STEEL TANKS PROGRAM

The Aboveground Steel Tanks Program manages the aging effect of the Aboveground Steel Tanks within the scope of License Renewal. This program includes preventive measures to mitigate corrosion and periodic inspections to manage the effects of loss of material due to corrosion on the exterior surface of the aboveground steel tanks within the scope of license renewal.

The program utilizes the application of a qualified protective coating on the exterior surface of the condensate storage tank to mitigate corrosion due to environmental factors. Inaccessible locations, such as the tank bottom are periodically monitored for material degradation using ultrasonic thickness measurements from the inside of the tank.

2013-015 This program is consistent with the ten elements of NUREG-1801 Rev. 1 XI.M29 and takes no exception to NUREG-1801 Rev. 1 XI.M29.

18.1.3 ASME SECTION XI, INSERVICE INSPECTION, SUBSECTIONS IWB, IWC, AND IWD PROGRAM

The ASME Section XI, Inservice Inspection, Subsections IWB, IWC, and IWD Program facilitates inspections to identify and correct degradation in Class 1, 2, and 3 piping components, supports, and integral attachments. The program includes periodic visual, surface, and /or volumetric examinations of all Class 1, 2, and 3 pressure-retaining components, supports, and integral attachments, including welds, pump casings, valve bodies, pressure-retaining bolting, and piping/component supports and leakage tests of pressure-retaining components.

The ASME Section XI, Inservice Inspection, Subsections IWB, IWC, and IWD Program manages the aging effect of cracking due to stress corrosion cracking, intergranular stress corrosion cracking, and irradiation assisted stress corrosion cracking. Duane Arnold has identified cracking in Class 1 large bore piping. The ASME Section XI, Inservice Inspection, Subsections IWB, IWC, and IWD Program detects any cracks which would result in the loss of fracture toughness due to thermal and neutron/radiation embrittlement.

2013-015 This program is consistent with the ten elements of NUREG-1801 Rev. 1 XI.M1 and takes no exception to NUREG-1801 Rev. 1 XI.M1.

18.1.4 ASME SECTION XI, INSERVICE INSPECTION, SUBSECTION IWE PROGRAM

The ASME Section XI, Inservice Inspection, Subsection IWE Program performs visual inspections, volumetric examinations and surface examinations in accordance with the ASME Code. The program manages aging effects for the drywell, suppression chamber, and connecting piping, supports and bolting. The airlocks and hatches are included with the drywell and suppression chamber.

2013-015 This program is consistent with the ten elements of NUREG-1801 Rev. 1 XI.S1 and takes no exception to NUREG-1801 Rev. 1 XI.S1.

18.1.5 ASME SECTION XI, INSERVICE INSPECTION, SUBSECTION IWF PROGRAM

The ASME Section XI, Inservice Inspection, Subsection IWF Program utilizes visual examinations in accordance with the ASME Code to determine the mechanical and structural condition of components and supports by verifying parameters such as clearances, settings, physical displacement, discontinuities and
2013-015 imperfections, such as loss of integrity of bolted or welded connections, loose or missing parts, debris, corrosion, erosion, or wear.

2013-015 This program is consistent with the ten elements of NUREG-1801 Rev. 1 XI.S3 and takes no exception to NUREG-1801 Rev. 1 XI.S3.

18.1.6 BOLTING INTEGRITY PROGRAM

The Bolting Integrity Program manages the aging effects associated with bolting through the performance of periodic inspections. The program includes repair/ replacement controls for ASME Section XI related bolting and generic guidance regarding material selection, thread lubrication and assembly of bolted joints. The program considers the guidelines delineated in NUREG-1339 for a bolting integrity program, EPRI NP-5769 (with the exceptions noted in NUREG-1339) for safety related bolting, and EPRI TR-104213 for non-safety related bolting. The Bolting Integrity Program credits five separate aging management programs for

the inspection of bolting. The five aging management programs are: (1) ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program, (2) ASME Section XI Inservice Inspection, Subsection IWF Program, (3) External Surfaces Monitoring Program, (4) Structural Monitoring Program, and (5) Buried
2013-015 Piping and Tanks Inspection Program.

2013-015 This program is consistent with the ten elements of NUREG-1801 Rev. 1 XI.M18 and takes no exception to NUREG-1801 Rev. 1 XI.M18.

18.1.7 BURIED PIPING AND TANKS INSPECTION PROGRAM

The Buried Piping and Tanks Inspection Program includes provisions for visual inspections of the protective wraps and coatings on buried carbon and low-alloy steel, piping and tanks and visual inspections of external surfaces of ductile iron, cast iron and stainless steel pipe in-scope for license renewal. The visual inspections for damage are performed when the carbon, low-alloy steel, ductile iron, cast iron and stainless steel components are excavated during maintenance and when a component is dug up and inspected for any reason. If damage to the protective wraps and coatings of carbon and low-alloy steel is found, the outer surface of the pipe or tank is inspected for loss of material. This DAEC program contains inspections for buried pipe that is not coated. Uncoated ductile iron, cast iron and stainless steel piping will be inspected for loss of material.

2013-007 Per Reference 4, buried piping inspections may be performed by a hydrostatic test or may be internally inspected by a method capable of precisely determining pipe wall thickness. These are alternatives to direct external inspections involving excavation. The alternatives will test at least 25% of the code class/safety-related or hazmat piping or both constructed from the material under consideration. The interval for these alternatives will not exceed 5 years.

2013-007 The Program is consistent with NUREG-1801 Rev. 1 XI.M34 with two exceptions. The Program expands the NUREG-1801 Rev. 1 XI.M34 requirements to include inspections for loss of material of uncoated ductile iron, cast iron and
2013-007 stainless steel pipes. The Program includes alternatives in lieu of direct external inspections involving excavation.

Cathodic Protection system availability will be maintained $\geq$ 90%. If 90% availability is not maintained, the condition will be entered into the corrective action program to evaluate the impact and take corrective actions. Each rectifier provides a single field of protection for buried piping and tanks for a total of five fields. Availability will be demonstrated by having no more than six months of rectifier out-of-service time in any sixty month period, as determined on a "per rectifier" basis; or no more than six months of rectifier out-of-service time in any twelve month period, for all rectifiers combined. Annual surveys will continue to be performed in accordance with NACE Standard Practice.

18.1.8 BWR CONTROL ROD DRIVE RETURN LINE NOZZLE PROGRAM

The BWR Control Rod Drive Return Line Nozzle Program ensures that cracks in the control rod drive return line nozzle due to thermal stress will be detected prior to loss of function. The program ensures that cracks in the control rod drive return line pipe containing stagnant water that is susceptible to intergranular stress corrosion cracking will be detected prior to loss of intended function.

Duane Arnold has removed the control rod drive return line nozzle thermal sleeve and installed a blind flange to prevent flow through the return line during plant operation to eliminate thermal cycling.

The ASME Section XI Inservice Inspection Program performs periodic ultrasonic inspections of the critical regions of the control rod drive return line nozzle. The Augmented Inspection Program inspects the control rod drive return line stainless steel pipe section welds that contain stagnant water and are susceptible to intergranular stress corrosion cracking.

2013-015 This program is consistent with NUREG-1801 Rev. 1 XI.M6 with two exceptions. The method for blocking the return line uses a blind flange instead of cutting and capping the line. The nozzle inspection frequency is based on ASME Code Section XI instead of NUREG-0619.

18.1.9 BWR FEEDWATER NOZZLE PROGRAM

The BWR Feedwater Nozzle Program consists of the ASME Section XI Inservice Inspection Program and the ASME Section XI Augmented Inspection Program as well as system modifications and operator instructions. The DAEC program performs feedwater nozzle inspections as required by ASME Section XI Subsection IWB, Table IWB 2500-1 (2001 edition including the 2002 and 2003 Addenda) and the recommendations of General Electric NE-523-A71-0594, Revision 1. The Augmented Inspection Program performs periodic ultrasonic inspection of critical regions of the Duane Arnold feedwater nozzle. The regions inspected, examination techniques, personnel qualifications, and inspection schedule are consistent with the recommendations of GE NE-523-A71-0594-A, Revision 1.

The feedwater nozzle design prevents flow of cold water behind the thermal sleeve which reduces the risk of cracking due to thermal cycling. Additionally, Duane Arnold has implemented changes to the controls of the feedwater regulating valves and placed cautions in operating procedures.

2013-015 This program is consistent with the ten elements of NUREG-1801 Rev. 1 XI.M5 and takes no exceptions to NUREG-1801 Rev. 1 XI.M5.

18.1.10 BWR PENETRATIONS PROGRAM

The BWR Penetrations Program is part of the ASME Section XI Inservice Inspection Subsection IWB, IWC, and IWD Program. The program utilizes ultrasonic (volumetric), surface and visual inspections. The program incorporates the guidelines of BWRVIP-49-A for instrument penetrations, and BWRVIP-27-A for the Standby Liquid Control System. Water chemistry is maintained and monitored by the Duane Arnold Water Chemistry Program.

2013-015 This program is consistent with the ten elements of NUREG-1801 Rev. 1 XI.M8 and takes no exception to NUREG-1801 Rev. 1 XI.M8.

18.1.11 BWR REACTOR WATER CLEANUP SYSTEM PROGRAM

The BWR Reactor Water Cleanup System Program ensures that cracks due to stress corrosion cracking and intergranular stress corrosion cracking in the Reactor Water Cleanup System pipe welds will be detected prior to loss of its intended function. The program includes periodic inspections, water chemistry control, and plant modifications.

2013-015 The program includes measures delineated in NUREG-0313, Rev. 2 and NRC Generic Letter 88-01, and Inservice Inspection in conformance with the American Society of Mechanical Engineers (ASME) Code, Section XI.

2013-015 The program is consistent with the ten elements of NUREG-1801 Rev. 1 XI.M25 and takes no exception to NUREG-1801 Rev. 1 XI.M25.

18.1.12 BWR STRESS CORROSION CRACKING PROGRAM

The BWR Stress Corrosion Cracking Program incorporates the guidelines of NRC Generic Letter 88-01 and Supplement 1, NUREG-0313 Rev. 2 and BWRVIP-75. The program has reduced the susceptibility to stress corrosion cracking by utilizing methods to reduce the tensile strength, such as: induction heating stress improvement, mechanical stress improvement process, weld overlay, or solution annealing. Water chemistry is maintained and monitored by the Water Chemistry Program.

2013-015 This program is consistent with the ten elements of NUREG-1801 Rev. 1 XI.M7 and takes no exception to NUREG-1801 Rev. 1 XI.M7.

18.1.13 BWR VESSEL ID ATTACHMENT WELDS PROGRAM

The BWR Vessel ID Attachment Welds Program utilizes portions of the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program. The program incorporates the guidelines of BWRVIP-48-A. Reactor water chemistry is maintained and monitored by the Water Chemistry Program.

2013-015 This program is consistent with the ten elements of NUREG-1801 Rev. 1 XI.M4 and takes no exception to NUREG-1801 Rev. 1 XI.M4.

18.1.14 BWR VESSEL INTERNALS PROGRAM

The BWR Vessel Internals Program utilizes applicable portions of the ASME Section XI Inservice Inspection Subsections IWB, IWC, and IWD Program, appropriate BWRVIP guidelines, and the Water Chemistry Program. The BWR Vessel Internals Program incorporates BWRVIPs: BWRVIP-18, BWRVIP-25, BWRVIP-26, BWRVIP-38, BWRVIP-41, BWRVIP-47, BWRVIP-76, and BWRVIP-139.

If any repairs are required as a result of the inspections performed by the above referenced BWRVIPs, the following BWRVIPs will be used, as applicable to implement the required repair: BWRVIP-02, BWRVIP-17, BWRVIP-16, BWRVIP-19, BWRVIP-44, BWRVIP-45, BWRVIP-50, BWRVIP-51, BWRVIP-52, BWRVIP-55-A, BWRVIP-57, and BWRVIP-58.

2013-007 2017-014 The BWR Vessel Internals Program will assess thermal and/or neutron embrittlement in susceptible Cast Austenitic Stainless Steel (CASS) components that are part of the reactor pressure vessel internals. The program will be implemented under the Augmented Inspection procedures for the BWR Vessel Internal Program rather than a standalone program. (See Section 18.1.38.) The visual inspections will be in accordance with applicable ASME Section XI and BWRVIP guidance.

2013-007 2017-014 This program is consistent with NUREG-1801 Rev. 1 XI.M9 with one exception. The Program includes the assessment for thermal and/or neutron embrittlement in susceptible CASS components.

18.1.15 CLOSED COOLING WATER SYSTEM PROGRAM

The Closed Cooling Water System Program relies on implementation of the
2013-015 guidance provided in EPRI 1007820 to ensure that the closed cycle cooling water system functions and components serviced by the system are not compromised by aging. The program includes control of chemistry parameters to minimize corrosion and stress corrosion cracking. DAEC performs testing and inspections of the CCCW systems, components to ensure the performance is maintained and the intended functions are not compromised by aging.

2013-015 This program is consistent with NUREG-1801 Rev. 1 XI.M21 with the exception that the program standard is EPRI 1007820 rather than EPRI TR-107396.

18.1.16 COMPRESSED AIR MONITORING PROGRAM

The Compressed Air Monitoring Program consists of inspection, monitoring, and testing of the compressed air systems (Safety Related Air, Instrument Air, Service Air, and Breathing Air), including (1) leak testing of valves, piping, and other system components, especially those made of steel and stainless steel; and (2) preventive monitoring that checks air quality at various locations in the system to ensure that oil, water, rust, dirt, and other contaminants are kept within the specified limits.

This program is in response to NRC GL 88-14 and INPO Significant Operating Experience Report (SOER) 88-01. It also relies on the ASME OM Guide Part 17, and ISA-S7.0.1-1996 as guidance for testing and monitoring air quality and moisture.

18.1.17 ELECTRICAL CABLES AND CONNECTIONS PROGRAM

The Electrical Cables and Connections Program manages the effects of aging by inspecting cables and connections susceptible to aging due to radiological, thermal and chemical aging mechanisms. Visual inspections will identify cables or connections degraded by these aging mechanisms.

Visually accessible cables and connections susceptible to thermal aging due to a combination of ambient temperature and ohmic heating will be inspected at least once every 10 years. If the cables and connections in these areas do not exhibit signs of aging then cables and connections in areas with lower ambient temperatures and ohmic heating will not exhibit signs of aging.

Cables and connections at equipment where significant heating can occur will be inspected. Equipment where significant heating can occur is defined as large motors (greater than 125 hp), motor operated valves, transformers, heaters, motor control centers, load centers, lighting panels and batteries.

Cables and connections in areas with elevated radiation levels will be visually inspected. Inspecting these cables and connections will provide reasonable assurance that cables and connections in areas with lower temperature and lower radiation dose rates will meet their intended functions.

This is a new program for Duane Arnold. New procedures and preplanned tasks will be developed and implemented to contain the scheduling information, instructions and acceptance criteria for the area inspections. Existing equipment maintenance procedures will be enhanced with steps for inspecting cables and connections during selected preventive maintenance activities and to document the cable inspection activity.

2013-015 This program is consistent with the ten elements of NUREG-1801 Rev. 1 XI.E1 and takes no exception to NUREG-1801 Rev. 1 XI.E1.

18.1.18 ELECTRICAL CABLES AND CONNECTIONS USED IN INSTRUMENTATION CIRCUITS PROGRAM

The Electrical Cables and Connections Used in Instrumentation Circuits Program manages the effects of aging by measuring the insulation resistance of the cables and connections at least once every 10 years. The test methodology is time domain reflectometry.

2013-015 This program is consistent with the ten elements of NUREG-1801 Rev. 1 XI.E2 and takes no exception to NUREG-1801 Rev. 1 XI.E2.

18.1.19 ELECTRICAL CONNECTIONS PROGRAM

The Electrical Connections Program manages the effects of aging by one-time inspection (thermographic) of a representative sample of electrical connections. The inspection is to validate that a periodic inspection program is or is not required to maintain the current licensing basis for the period of extended operation. The one-time inspection will provide the basis needed to conclude that an aging management program for electrical connections is or is not required.

2013-015 This is site specific program. There is no NUREG-1801 Rev. 1 Volume 2 Section XI program that matches this program.

18.1.20 ELECTRICAL PENETRATION ASSEMBLIES PROGRAM

The Electrical Penetration Assemblies Program is a plant-specific program that manages the effects of aging by inspecting the electrical penetration assemblies periodically. This aging management program is identical to the required maintenance activities for the electrical penetration assemblies within the scope of the Environmental Qualification Program.

Duane Arnold has experienced the failure of two electrical penetration assemblies. An analysis of one concluded that the failure was due to moisture, a random void, and a potential difference between conductors with subsequent growth of dendrites between the conductors. The dendrites formed a low resistance path, over a long period of time, for current leakage, arching, and carbonization of the epoxy. The electrical short developed when the carbonized path between the conductors became continuous and resulted in shorting between the splices of the two conductors. The moisture could have been due to less than adequate adherence to manufacturer's instructions which required internal nitrogen pressure be maintained in the assemblies. Dendrites formation required the presence of moisture.

2013-015 This is site specific program. There is no NUREG-1801 Rev. 1 Volume 2 Section XI program that matches this program.

18.1.21 EXTERNAL SURFACES MONITORING PROGRAM

The External Surfaces Monitoring Program manages aging effects of loss of material using visual inspection of external surfaces. The program consists of
2013-015 periodic inspections of metallic and polymeric components such as piping, piping components, ducting, pipe supports, and other components within the scope of license renewal.

2013-015 This program is consistent with the ten elements of NUREG-1801 Rev. 2 XI.M36 and takes no exception to NUREG-1801 Rev. 2 XI.M36.

18.1.22 FIRE PROTECTION PROGRAM

The Fire Protection Program manages aging effects of fire protection components using surveillance test procedures and detailed inspections. Surveillance tests are performed on the diesel-driven fire pump, the CO_2 fire suppression system, fire doors, and fire barrier penetration seals. Visual inspections for degradation are performed on fire barrier walls, ceilings and floors.

2013-015 This program is consistent with NUREG-1801 Rev. 1 XI.M26 with the following two exceptions to NUREG-1801 Rev. 1 XI.M26:

- Inspections of 35 percent of fire barriers, walls, ceilings and floors will be conducted every 18 months with 100 per cent visually inspected within five years. NUREG-180 XI.M26 recommends that these inspections be performed every refueling cycle.

- The CO_2 Cardox System Operability Test procedure examines the CO_2 fire suppression system for the cable spreading room annually for signs of degradation (e.g., corrosion, mechanical damage, or damage to dampers). NUREG-1801 XI.M26 recommends inspection every six months.

18.1.23 FIRE WATER SYSTEM PROGRAM

The Fire Water System Program manages aging effects of fire protection components using surveillance test procedures and detailed inspections. Fire Water System components are tested in accordance with the applicable National Fire Protection Association (NFPA) codes and standards.

2013-015 This program is consistent with the ten elements of NUREG-1801 Rev. 1 XI.M27 and takes no exception to NUREG-1801 Rev. 1 XI.M27.

18.1.24 FLOW ACCELERATED CORROSION PROGRAM

The Flow Accelerated Corrosion Program manages aging effect of loss of material due to flow-accelerated corrosion (FAC) on the internal surfaces of carbon or low alloy steel piping, elbows, reducers, tees, expanders, and valve bodies which contain high energy fluids (both single phase and two phase flow). The program is based on the guidelines of NSAC-202L-R3. This program uses CHECWORKS as a predictive tool. Included in the program are: (a) an analysis to determine flow-accelerated corrosion susceptible lines; (b) performance of baseline inspections; (c)

follow-up inspections to confirm the predictions; and (d) repairing or replacing components, as necessary.

2013-015 This program is consistent with NUREG-1801 Rev. 1 XI.M17 with one exception. The DAEC program implements the guidance provided in EPRI-NSAC-202L-R3, "Recommendations for an Effective Flow-Accelerated Corrosion Program," May 2006, in lieu of the NUREG-1801 recommendation of EPRI NSAC-202L-R2, "Recommendations for an Effective Flow-Accelerated Corrosion Program," April 1999.

18.1.25 FUEL OIL CHEMISTRY PROGRAM

The Fuel Oil Chemistry Program complies with the plant Technical Specifications. The program consists of surveillance test procedures with supporting maintenance and chemistry procedures. The periodicity of surveillance tests allow sufficient time to correct high particulate levels prior to reaching the limit of acceptability.

2013-015 This program is consistent with NUREG-1801 Rev. 1 XI.M30 with the following exceptions:

- NUREG-1801 states: Fuel oil quality is maintained by monitoring and controlling fuel oil contamination in accordance with the plant's technical specifications and the guidelines of the American Society for Testing Materials (ASTM) Standards D 1796, D2276, D 2709, D 6217, and D 4057. For determination of particulates the ASTM D 6217 or Modified ASTM D 2276, Method A is recommended. The DAEC Fuel Oil Chemistry Program does not use ASTM D 6217. DAEC uses the non-modified ASTM D 2276 which uses the more conservative filter pore size of 0.8µm verses the 3.0µm as used by the Modified ASTM D 2276, Method A. The DAEC Operating Experience and generally the industry Operating Experience shows this to be acceptable.

- DAEC does not use fuel additives of biocides to minimize biological activity, stabilizers to prevent biological breakdown of the diesel fuel, nor corrosion inhibitors to mitigate corrosion. The monthly testing for and removal of water and the purchase of quality fuel oil negates the need for additives. The DAEC Operating Experience shows this to be an acceptable alternative.

18.1.26 FUSE HOLDERS PROGRAM

The Fuse Holders Program manages the effects of aging by visual and thermographic inspection. The visual inspection is to identify aging due to adverse localized environments.

The thermographic inspection is to identify aging due to loosening of the metal clip. Plant procedures contain the scheduling information, instructions and acceptance criteria for performing thermography on control panels.
Fuse holders within the scope of license renewal will be inspected at least once every 10 years. The first inspection is to be completed before the period of extended operation.

2013-015 This program is consistent with NUREG-1801 Rev. 1 XI.E5 with the following exception:

- The program takes exception to the following aging mechanisms listed in NUREG-1801 XI.E5:

  Electrical Transients
  The only electrical transients significant enough to cause fatigue are phase to ground, phase to phase or three phase faults. These electrical transients are events and not aging mechanisms.

  Vibration
  Fuse holders are installed in panels. Panels are not sources of vibration and are installed to minimize vibrations being transmitted to equipment in the panel.

  Chemical Contamination
  Plant design and installation practices provide appropriate protection for fuse holders from chemical contamination by requiring fuses to be installed in enclosures. Boric acid chemical contamination is not a concern for boiling water reactors.

  Corrosion
  Plant installation and maintenance practices provide appropriate protection for fuse holders from moisture intrusion (such as in enclosures). The location of fuse holders was reviewed to identify fuse holders installed outside of an active device, junction box, or similar type enclosures (i.e., unprotected environment). This review identified no unprotected fuses. Boric acid chemical contamination is not a concern for boiling water reactors. Panels protect the fuse holders from the causes of corrosion (moisture and chemicals).

  Oxidation
  Oxidation is not an aging mechanism unless there are other chemicals or moisture present. The panels protect the fuse holder from chemicals and moisture.

2013-005 18.1.27 INACCESSIBLE CABLES PROGRAM

2013-005 The Inaccessible Cables Program manages the effects of aging by measuring the insulation resistance of the cables and connections at least once every 10 years in 2013-005 accordance with plant procedures. In-scope cables exposed to significant moisture are tested to provide an indication of the condition of the conductor insulation. Significant moisture is defined as periodic exposures to moisture that lasts more than a few days (e.g., cable wetting or submerged in water). A proven test for detecting deterioration of the insulation system for all power cables (480 V to 35 kV) due to wetting, such as power factor, partial discharge, or polarization index, as described in EPRI TR-103834-P1-2, or other testing that is state-of-the-art and commercially available at the time the test is performed.

2013-005 This aging management program includes actions to prevent cables from being exposed to long term significant moisture by inspecting the manholes and conduits containing cables and testing of sump pumps at least once every 2 years. Actual frequency is based on operating experience. These activities are controlled by a preplanned task.

2013-015 This program is consistent with the ten elements of NUREG-1801 Rev. 1 2013-005 XI.E3 and takes one exception to NUREG-1801 Rev. 1 XI.E3. The scope of the program is expanded to include all cables that support a license renewal or maintenance rule function. The program is enhanced to include inaccessible 480 V to 2 kV power cables and to include 2 kV to 35 kV cables that are not energized greater than 25% of the time.

18.1.28 INSPECTION OF INTERNAL SURFACES IN MISCELLANEOUS PIPING AND DUCTING COMPONENTS PROGRAM

The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting 2013-015 Components Program consists of inspections of the internal surfaces of metallic piping, piping components, ducting, polymeric components and other components not covered by other aging management programs. The program consists of visual inspections performed during pre-planned system and component maintenance activities when the systems are opened and the surfaces are accessible.

2013-015 This program is consistent with the ten elements of NUREG-1801 Rev. 2 XI.M38 and takes no exception to NUREG-1801 Rev. 2 XI.M38.

18.1.29 INSPECTION OF OVERHEAD HEAVY LOAD AND LIGHT LOAD (RELATED TO REFUELING) HANDLING SYSTEMS PROGRAM

The Inspection of Overhead Heavy Load and Light Load (related to refueling) Handling Systems Program manages loss of material of structural components of heavy and light load handling systems. The Program addresses loss of material due to general corrosion of supporting steel and loss of material due to wear on the crane rails through periodic visual inspection.

2013-017 In addition, the Program tracks over-capacity lifts for the reactor building crane and turbine building crane.

Inspection of the torus monorail is completed as part of the Technical Specification Surveillance for the suppression chamber and drywell visual inspection.
2013-015 This program is consistent with the ten elements of NUREG-1801 Rev. 1 XI.M23 and takes no exception to NUREG-1801 Rev. 1 XI.M23.

18.1.30 LUBRICATING OIL ANALYSIS PROGRAM

The Lubricating Oil Analysis Program ensures the oil environment in the mechanical systems is maintained to the required quality. The program maintains oil contaminants (primarily water and particulates) within acceptable limits to manage the aging effects of loss of material, cracking, and heat transfer degradation. Oil testing activities include periodic sampling, analysis, and trending of results.

2013-015 This program is consistent with the ten elements of NUREG-1801 Rev. 1 XI.M39 and takes no exception to NUREG-1801 Rev. 1 XI.M39.

18.1.31 METAL ENCLOSED BUS PROGRAM

The Metal Enclosed Bus Program manages the effects of aging by inspecting the insulation of the metal enclosed bus periodically. Maintenance procedures and preplanned tasks contain the scheduling information, instructions and acceptance criteria for inspecting the metal enclosed bus within the scope of this program.

The Duane Arnold program applies to buses that support a license renewal function and are susceptible to any of the following aging mechanisms:

- Loosening of bolted connections due to thermal cycling and ohmic heating
- Reduced insulation resistance
- Moisture/debris intrusion

The non-segregated buses between the startup transformer and the 4.16 kV switchgear are metal enclosed buses within the scope of this program.
2013-015 This program is consistent with NUREG-1801 Rev. 1 XI.E4 with the following exception:

- NUREG-1801 XI.E4 recommends a 5 year frequency for visual inspections when no thermographic inspections are performed. The DAEC performs the visual inspections on a 6 year frequency as part of the major inspection of the associated transformer. The inspections that have been performed since the bus bar insulation was replaced have not identified any degradation. Therefore, performing visual inspections on a 6 year frequency provides reasonable assurance that the metal enclosed bus will be maintained consistent with the current licensing basis through the period of extended operation.

18.1.32 ONE-TIME INSPECTION PROGRAM

The One-Time Inspection Program addresses potentially long incubation periods for certain aging effects and provides a means of verifying that an aging effect is either not occurring or progressing so slowly as to not have an effect on the intended function of the structure or component. The program provides measures for
2013-007 verifying an aging management program is not needed, verifying the effectiveness of other aging management programs, or determining that degradation is occurring which will require evaluation and corrective action.

The program assesses loss of material due to crevice, galvanic, general, pitting, and microbiologically-influenced corrosion and erosion, heat transfer degradation due to fouling, and cracking due to stress corrosion cracking or cyclic loading.

2013-007 The sample selection for the DAEC One-Time Inspection program will include a representative sample of the population. Existing maintenance records that document component condition will be used as part of the sample. (See Reference 4.)

2013-007 This program is consistent with the ten elements of NUREG-1801 Rev. 1 XI.M32 and takes no exception to NUREG-1801 Rev. 1 XI.M32.

18.1.33 OPEN CYCLE COOLING WATER SYSTEM PROGRAM

The Open Cycle Cooling Water System Program relies on implementation of NRC Generic Letter 89-13 to ensure that the effects of aging on the raw water systems are managed for the period of extended operation.

The OCCW program manages the aging effects in the following systems:

- Circulating Water System
- River Water Supply System
- Residual Heat Removal Service Water System
- Emergency Service Water System

2013-015 This program is consistent with NUREG-1801 Rev. 1 XI.M20 with one exception taken to the requirement for metal surfaces of underlying system components to be lined or coated. The DAEC open cycle cooling water (OCCW) piping included within the scope of this program is constructed of carbon steel that is not lined or coated. The original design of the DAEC piping for OCCW systems selected unlined/uncoated piping that is acceptable for the environment and intended functions of these piping systems.

18.1.34 REACTOR HEAD CLOSURE STUDS PROGRAM

The Reactor Head Closure Studs Program is an integral part of the ASME Section XI Inservice Inspection Program. The program incorporates the appropriate Code edition and sections of ASME Section XI Subsection IWB. The program provides preventive measures to mitigate cracking. These measures include material selection, appropriate coatings, and lubrications which follow the guidelines of NRC Regulatory Guide 1.65.

2013-015 This program takes exception to the NUREG-1801 Rev. 1 XI.M3 requirement to perform surface and volumetric inspections of studs when removed. DAEC inspection of the reactor head closure studs program is performed in accordance with the applicable portions of ASME Section XI and 10 CFR 50.55a which do not necessarily require both inspections.

18.1.35 REACTOR VESSEL SURVEILLANCE PROGRAM

The Reactor Vessel Surveillance Program is consistent with the requirements
2013-015 of 10 CFR 50, Appendix H, NRC Regulatory Guide (RG) 1.99 and ASTM E-185. The program manages the effects of neutron/radiation embrittlement on the reactor pressure vessel beltline.

2013-015 This program is consistent with the ten elements of NUREG-1801 Rev. 1 XI.M31 and takes no exception to NUREG-1801 Rev. 1 XI.M31.

18.1.36 SELECTIVE LEACHING OF MATERIALS PROGRAM

The Selective Leaching of Materials Program will ensure that cast iron, brass, bronze, and copper alloy components exposed to raw water, treated water, or ground water will maintain their integrity for the period of extended operation. The program will include a one-time visual inspection and mechanical test of selected components that may be susceptible to selective leaching.

2013-015 The program is consistent with NUREG-1801 Rev. 1 XI.M33 with one exception. Visual inspection and mechanical test techniques (Brinnell hardness testing or other mechanical tests such as mechanical scraping, chipping or other types of hardness testing), or additional examination methods that become available to the nuclear industry, are used to determine if selective leaching is occurring on the surfaces of a selected set of components. The GALL recommends that visual inspections be performed with Brinnell hardness testing.

18.1.37 STRUCTURES MONITORING PROGRAM

The Structures Monitoring Program includes the Masonry Wall Program and the Inspection of Water Control Structures Associated with Nuclear Power Plants Program.

The Structures Monitoring Program includes periodic visual inspection of structures and structural components for the detection of aging effects. Detection of aggressive subsurface environments will be completed by periodic sampling the on-site groundwater. Examinations of inaccessible areas, such as buried concrete foundations, will be completed during inspections of opportunities during pre-planned maintenance activities. The Masonry Wall Program includes visual inspection of safety-related masonry walls for degradation. Periodic visual inspections of water controlled structures associated with the emergency core cooling water systems and/or flood protection are conducted in accordance with the Maintenance Rule Program. Individuals performing inspections and reviews will be qualified in accordance with the Maintenance Rule Program.

2013-015 This program is consistent with the ten elements of NUREG-1801 Rev. 1 XI.S5, XI.S6 and XI.S7 and takes no exception to NUREG-1801 Rev. 1 XI.S5, XI.S6 and XI.S7.

18.1.38 THERMAL AGING AND NEUTRON IRRADIATION EMBRITTLEMENT OF CAST AUSTENITIC STAINLESS STEEL (CASS) PROGRAM

2013-007 2017-014 Per Reference 4, a stand-alone Thermal Aging and Neutron Irradiation Embrittlement of CASS Program will not be implemented, as the assessment is performed under another aging management program, BWR Vessel Internals Program. (See Section 18.1.14.)

18.1.39 WATER CHEMISTRY PROGRAM

The Water Chemistry Program establishes the plant water chemistry specifications, action levels, and responses to out-of-specification water chemistry conditions. The program relies on monitoring and control of reactor water chemistry
2012-003 based on industry guidelines of BWRVIP-190.

2013-015 This program is consistent with the ten elements of NUREG-1801 Rev. 1 XI.M2 and takes no exception to NUREG-1801 Rev. 1 XI.M2.

18.1.40 ASME CODE CLASS 1 SMALL-BORE PIPING INSPECTION PROGRAM

The ASME Code Class 1 Small-Bore Piping Inspection Program is a plant-specific program that manages cracking of small-bore class 1 piping. Ten percent of Class 1 butt welds in piping of less than four inch NPS receive a volumetric examination each interval. Ten percent of Class 1 socket welds in piping of less than four inch NPS receive a volumetric examination each interval. A volumetric technique for ASME Code Class 1 small-bore socket welds that is endorsed by the industry is used, if available. If no such technique is available at the time the inspections are performed, then a plant procedure for volumetric examination of ASME Code Class 1 small-bore piping with socket welds will be used. Socket welds less than four inch NPS receive a VT-2 visual examination during pressure testing during each refueling outage.

2013-015 A destructive examination may be performed on an opportunistic basis in lieu of the socket weld volumetric examinations.

18.1.41 BORAL SURVEILLANCE PROGRAM

The Boral Surveillance Program provides representative coupon testing for Holtec spent fuel racks and in-situ neutron attenuation testing for PaR spent fuel racks in order to ensure the Boral in the spent fuel racks continues to meet the assumptions of the spent fuel pool criticality analysis.

18.2 TLAA EVALUATION OF AGING MANAGEMENT PROGRAMS UNDER 10 CFR 54.21(C)(1)(iii)

18.2.1 ENVIRONMENTAL QUALIFICATION PROGRAM

The Duane Arnold Environmental Qualification Program ensures that the electrical components important to safety meet the requirements of 10 CFR 50.49. Station procedures identify components that are managed by this program.

2013-015 The program is consistent with the ten elements of NUREG-1801 Rev. 1 X.E1 and takes no exception to NUREG-1801 Rev. 1 X.E1.

2013-015 18.2.2 METAL FATIGUE OF REACTOR VESSEL COOLANT PRESSURE BOUNDARY PROGRAM

The Duane Arnold Metal Fatigue of Reactor Coolant Pressure Boundary Program is an existing program. In accordance with NUREG/CR-6260, the program has evaluated the impact of environmental effects on fatigue usage and shown them to be less than the maximum allowable (1.0) for the period of extended operation.

The following components are evaluated:

- Reactor pressure vessel shell and lower head
- Reactor pressure vessel recirculation outlet nozzle
- Reactor pressure vessel recirculation inlet nozzle
- Reactor pressure vessel feedwater nozzle bore
- Reactor pressure vessel core spray nozzle and safe end
- Feedwater/reactor core isolation cooling tee
- Recirculation piping/residual heat removal return tee
- ASME Class 1 residual heat removal piping at tapered transition

2013-015 This program is consistent with the ten elements of NUREG-1801 Rev. 1 XI.M1 and takes no exception to NUREG-1801 Rev. 1 XI.M1.

18.2.3 EXEMPTIONS

The requirements of 10 CFR 54.21(c) stipulate that the application for a renewed license should include a list of plant-specific exemptions granted pursuant to 10 CFR 50.12 and that are based on time-limited aging analyses, as defined in 10 CFR 54.3. Each active 10 CFR 50.12 exemption has been reviewed to determine whether the exemption is based on a time-limited aging analysis. No existing TLAA related exemptions were identified.

18.3 TIME-LIMITED AGING ANALYSES

As part of a license renewal application, 10 CFR 54.21(c) requires that an evaluation of TLAAs for the period of extended operation be provided. The following TLAAs have been identified for Duane Arnold and evaluated to meet this requirement.

18.3.1 NEUTRON EMBRITTLEMENT OF THE REACTOR PRESSURE VESSEL

The materials of the RPV and internals are subject to embrittlement due to high energy (E > 1 MeV) neutron exposure. Embrittlement means the material has lower toughness (i.e., will absorb less strain energy during a crack or rupture), thus allowing a crack to propagate more easily under thermal and/or pressure loading.
The reactor vessel neutron embrittlement TLAAs have been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii). Fifty-four effective full power years (EFPY) would be the effective full power years at the end of the period of extended operation assuming an average capacity factor of 90% for 60 years.

Analyses were performed to determine neutron fluence for extended operation to 54 EFPY. High energy (>1 MeV) neutron fluence for the welds and shells of the RPV beltline region was calculated using the RAMA fluence methodology. The RAMA methodology was developed for the Electric Power Research Institute and the Boiling Water Reactor Vessel and Internals Project. Use of this methodology for evaluations of fluence for the DAEC was performed in accordance with guidelines presented in Regulatory Guide 1.190, as recommended in NUREG-1800. The NRC has reviewed and approved RAMA for BWR RPV fluence predictions.

18.3.1.1 REACTOR VESSEL UPPER SHELF ENERGY REDUCTION

Upper Shelf Energy (USE) is the standard industry parameter used to indicate the maximum toughness of a material at high temperature. 10CFR50 Appendix G requires the predicted end-of-life Charpy impact test USE for RPV materials to be at least 50 ft-lb (absorbed energy), unless an approved analysis supports a lower value. The predicted USE drop is determined in accordance with NRC Regulatory Guide 1.99, Revision 2. For Boiling Water Reactors (BWRs) that cannot meet the 50 ft-lb criterion, the BWR Vessel and Internals Project (BWRVIP) has provided a bounding equivalent margins USE analysis for plants in BWRVIP-74-A, which is valid for up to 54 EFPY of operation.

Predicted USE drop for each RPV material in the beltline region exposed to fluence greater than 1.0×10^{17} n/cm^2 for 54 EFPY was determined in accordance with RG 1.99. In cases where the 50 ft-lb criterion cannot be met, or where USE data is absent, an equivalent margin analysis (EMA) using BWRVIP-74-A was performed.

All DAEC materials are acceptable from a USE standpoint for 54 EFPY.

18.3.1.2 ADJUSTED REFERENCE TEMPERATURE INCREASE

The adjusted reference temperature (ART) of the limiting beltline material is used to adjust the beltline P-T curves to account for irradiation effects.

The DAEC ART values were determined in accordance with Regulatory Guide (RG) 1.99, Revision 2. The limiting beltline plate material has an ART value of 152.6°F for 54 EFPY. The limiting nozzle, N-2, has an ART value of 116.4°F for 54 EFPY. ART values for 54 EFPY are below the 200ºF suggested in Regulatory Guide 1.99 and are, therefore, acceptable for the period of extended operation.

18.3.1.3 REACTOR VESSEL THERMAL LIMIT - OPERATING PRESSURE - TEMPERATURE LIMITS

Revised P/T curves were created for 54 effective full power years (EFPY) of operation, using the methodology of the 2001 Edition, 2003 Addenda of ASME Code, Section XI, Appendix G, and 10CFR50 Appendix G. The curves were developed in accordance with the methodology of the Boiling Water Reactor Owners' Group (BWROG) Licensing Topical Report, "Pressure Temperature Limits Report Methodology for Boiling Water Reactors" Structural Integrity Associates Report No. SIR-05-044-A, Revision 0, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," April 2007). Fluence was determined using NRC-approved RAMA methodology.

18.3.1.4 REACTOR VESSEL CIRCUMFERENTIAL WELD EXAMINATION RELIEF

Relief from RPV circumferential weld examination requirements under GL 98-05 is based on probabilistic assessments that predict an acceptably low probability of failure per reactor operating year. The analysis is based on RPV metallurgical conditions as well as flaw indication sizes and frequencies of occurrence that are expected at the end of a licensed operating period. The anticipated changes in metallurgical conditions expected over the extended licensed operating period require an additional analysis for 54 EFPY and approval by the NRC to extend this relief request.

An evaluation was performed based on the methodology presented in EPRI Report No. TR-105697, "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05)." The evaluation included the estimate of the probability of failure due to a limiting event (i.e., low temperature over-pressurization, or LTOP) for the case of 90% axial weld inspection (based on actual weld inspection coverage achieved in previous examinations). Probability of failure (PoF) results were calculated for 60 years (54 EFPY) for the RPV beltline axial welds and the beltline circumferential weld, including the consideration of the LTOP occurrence probability of $1x10^{-3}$ per year. The probability of failure for the circumferential welds is below that calculated in the Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report (TAC No. M93925), dated July, 1998.

18.3.1.5 REACTOR VESSEL AXIAL WELD PROBABILITY OF FAILURE

The DAEC inspection coverage on axial welds satisfies ASME Code requirements. The probability of failure per reactor year for the axial welds is below the probability quoted in the Supplement to Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report (TAC No. MA3395) dated March 7, 2000.

18.3.1.6 REFLOOD THERMAL SHOCK OF THE REACTOR VESSEL

A recent analysis performed for BWR-6 vessels evaluates the bounding LOCA event, a main steam line break, for a BWR vessel design that is similar to the DAEC vessel. Since the DAEC vessel inside diameter is appreciably smaller than the BWR-6 vessel sizes evaluated in the BWR-6 analysis, and the DAEC vessel beltline has a wall thickness less than that evaluated in the BWR-6 analysis, the cooldown due to the reflood event at the 1/4T depth would potentially be greater for the DAEC vessel than that of the BWR-6 vessel. A re-evaluation was performed for the DAEC which determined that the bounding applied stress intensity factor, K, for DAEC of 100 ksi√inch is less than the available fracture toughness of 200 ksi√inch after 54 EFPY, which is acceptable.

18.3.1.7 REACTOR INTERNALS

Irradiation Assisted Stress Corrosion Cracking

Austenitic stainless steel RPV internal components exposed to neutron fluence greater than 5 $x10^{20}$ n/cm^2 (E > 1 MeV) are considered susceptible to IASCC in the BWR environment; IASCC of RPV internals is considered a TLAA.

Therefore, IASCC of the following is a TLAA for the DAEC:

- Shroud
- Top guide
- Core support plate
- Incore instrumentation dry tubes and guide tubes

Core Plate Rim Hold-Down Bolts

As described in the SER to BWRVIP-25, plants must consider relaxation of the rim hold-down bolts as a TLAA issue.

For the DAEC, evaluation shows that at the end of plant life, the loss-of-preload caused by cracking in the rim hold-down bolts would not diminish the integrity of the core plate. The amount of preload lost after sixty years would be only 53 pounds of the original 10,980 pounds. Even if all the hold-down bolts cracked to 30% of their radii, the loss-of-preload would not diminish core plate integrity.

18.3.2 METAL FATIGUE

Fatigue is the progressive localized permanent structural change that occurs in a material subjected to repeated or fluctuating strains at nominal stresses having maximum values often much less than the tensile strength of the material. In the case of the Duane Arnold reactor pressure vessel, fatigue is based on the postulated cycles during operation of the plant; the most common of these being the startup/shutdown cycle. To address this design consideration for the reactor pressure vessel, explicit metal fatigue calculations were specified in the ASME Boiler and Pressure Vessel Code.

18.3.2.1 REACTOR PRESSURE VESSEL FATIGUE

The cumulative usage factor (CUF) values obtained from the 40 year analyses were updated to incorporate revised numbers of cycles for sixty years of operation. As shown by the analysis, the 60 year CUFs are less than 1.0, and therefore are acceptable.

18.3.2.2 REACTOR VESSEL INTERNALS FATIGUE

No plant specific fatigue analysis of the entire reactor vessel internals was performed.

18.3.2.3 FATIGUE OF CLASS 1, 2 AND 3 PIPING

Class 1 Piping

DAEC Class 1 piping systems were designed in accordance with B31.1 or B31.7 requirements. Those piping systems designed in accordance with B31.7 were explicitly analyzed for fatigue. These B31.7 evaluations have been reviewed to ensure that CUFs will remain less than 1.0 for 60 years of operation, or that the fatigue exemptions remain valid.

For the systems that were designed in accordance with ANSI B31.1 methodology, fatigue usage factors were not determined. For these systems, although the code of construction did not invoke fatigue analyses, a stress range reduction factor which is applied to the allowable stress range for expansion stresses is required to account for cyclic thermal conditions. The stress range reduction factor is 1.0 for 7,000 equivalent full temperature thermal cycles (or less). Since this piping will not exceed 7000 full temperature cycles in 60 years of operation, stress analyses remain valid for the period of extended operation.

Class 2 and 3 Piping

For Class 2 and 3 piping systems designed in accordance with B31.1 or B31.7, no explicit analysis for fatigue was required by the Code. For these systems, a stress range reduction factor which is applied to the allowable stress range for expansion stresses is required to account for cyclic thermal conditions. The stress range reduction factor is 1.0 for 7,000 equivalent full temperature thermal cycles (or less). Since this piping will not exceed 7000 full temperature cycles in 60 years of operation, existing stress analyses for non-Class 1 piping remain valid for the period of extended operation.

18.3.2.4 EFFECTS OF REACTOR COOLANT ENVIRONMENT (GSI-190)

2013-015 Generic Safety Issue (GSI) 190, "Fatigue Evaluation of Metal Components for 60-year Plant Life," was identified by the NRC because of concerns about the effects of reactor water environments on the fatigue life of components and piping during the period of extended operation. GSI-190 was closed in December of 1999, and concluded that environmental effects have a negligible impact on core damage frequency, and as such, no generic regulatory action is required. However, as part of the closure of GSI-190, the NRC concluded that licensees who apply for license renewal should address the effects of coolant environment on component fatigue life as part of their aging management programs.

Detailed environmental fatigue calculations were performed for DAEC for locations associated with the older vintage GE plant discussed in NUREG/CR-6260. Per Section X.M1 of the GALL Report, the EAF evaluation must use the appropriate F_{en} relationships from NUREG/CR-6583 (for carbon/low alloy steels) and NUREG/CR-5704 (for stainless steels), as appropriate for the material for each location. The methodology documented in NUREG/CR-6583 and NUREG/CR-5704 was used to evaluate environmental effects for DAEC components. To perform the environmental fatigue evaluations, HWC conditions were assumed to exist for 72.4% of the time, and NWC conditions to exist for 27.6% of the time.

The cumulative usage factors, including environmental effects, are shown to be below 1.0.

18.3.3 ENVIRONMENTAL QUALIFICATION

18.3.3.1 ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT (EQ)

The Duane Arnold Environmental Qualification Program was designed to meet Code of Federal Regulations, Title 10, Section 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants."

10CFR50.49(a) states, "Each holder of or each applicant for a license to operate a Nuclear Power Plant shall establish a program for qualifying ..." electric equipment as defined in the code.

(a) Electric equipment as defined by 10CFR50.49(b) shall be identified and if not located in a mild environment as defined by 10CFR50.49(c)(3) shall be included in the EQ Program.

(b) A master list of equipment in the EQ Program shall be prepared and maintained in accordance with 10CFR50.49 (d).

(c) Performance specifications, electrical characteristics and environmental conditions as defined in 10CFR50.49 (d), shall be established and maintained for equipment in the EQ Program in a qualification file.

(d) Qualification requirements and methods of qualification defined in 10CFR50.49 (e) and (f) shall establish the basis for the qualification of equipment in the EQ Program.

(e) In accordance with 10CFR50.49 (j), a record file is maintained, in an auditable form, containing information permitting verification that EQ equipment:

- "is qualified for its application"
- "meets its specified performance requirements when it is subjected to the condition predicted to be present when it must perform its safety function up to the end of its qualified life"
- shall be established and maintained for the entire period the equipment is installed or stored for future use at the plant.

(f) 10CFR50.49 (k) permits the continued environmental qualification of equipment qualified, in accordance with "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors", November 1979 (DOR Guidelines), prior to the issuance of 10CFR50.49.

(g) EQ equipment or components, replaced during plant modification or maintenance, shall be qualified as required by 10CFR50.49 (l) unless there are sound reasons to the contrary.

- These "sound reasons" shall be those delineated in Regulatory Guide 1.89, Rev. 1.

As required in Section 7.0 of DOR Guidelines, an ongoing program of surveillance and maintenance to assure that EQ equipment exhibiting age-related degradation will be identified and replaced as necessary, shall be established.

EQ Program documentation and equipment shall meet the applicable quality assurance requirements defined in 10CFR50 Appendix B.

In general, EQ components are qualified via simulated aging and testing to specified conditions in accordance with accepted regulatory requirements and industry standards. A qualified life for each component may be determined based on the test results in a number of ways, often using activation energies of each material in conjunction with the Arrhenius equation for thermal effects, and total accumulated dose respectively.

Reanalysis of an aging evaluation to extend the qualification of a components is performed on a routine basis pursuant to 10 CFR 50.49(e) as part of the EQ Program. While a component life limiting condition may be due to thermal or radiation aging, the 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 in Tab E of each EQR file in accordance with the requirements of the DAEC quality assurance (QA) program, which requires 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 below.

Analytical Methods – The EQ Program uses the same analytical models in the reanalysis of an aging evaluation as those previously applied during the prior evaluation. The Arrhenius methodology is an acceptable model for performing a thermal aging evaluation. The analytical method used for a radiation aging evaluation is to demonstrate qualification for the total integrated dose (that is, normal radiation dose for the projected installed life plus accident radiation dose). For license renewal, acceptable methods for establishing the 60 year normal radiation dose include multiplying the 40 year normal radiation dose by 1.5 (that is 60 years/40 years) or using the actual calculated value for 60 years. The result is added to the accident radiation dose to obtain the total integrated dose for the component. In many cases, the normal radiation dose is insignificant when compared to the accident dose. In such cases, the use of the accident dose is valid for both the 40 year and 60 year dose.

Data Collection and Reduction Methods – Reducing excess conservatism in the component service conditions (e.g., temperature, radiation) used in the prior aging evaluation is the main method used for a reanalysis per the EQ Program. Temperature data used in an aging evaluation should be conservative and based on plant design temperatures or on actual plant temperature data. When used, plant temperature data can be obtained in several ways including monitors used for technical specification compliance, other installed monitors, measurements made by plant operators during rounds, and temperature sensors on large motors (while the motor is not running). A representative number of temperature measurements are conservatively evaluated to establish the temperatures used in an aging evaluation. Plant temperature data may be used in an aging evaluation in different ways, such as (a) directly applying the plant temperature data in the evaluation or (b) using the plant temperature data to demonstrate conservatism when using plant design temperatures for an evaluation. Any changes to material activation energy values as part of a reanalysis are to be justified on a plant-specific basis.

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.

Acceptance Criteria and Corrective Action – Under the EQ Program, the reanalysis of an aging evaluation could extend the qualification of the component. If the qualification cannot be extended by reanalysis, the component must be refurbished, replaced, or re-qualified 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 re-qualify the component if the reanalysis is unsuccessful).

Based on a review of the DAEC EQ Program and operating experience, the continued effective implementation of the program provides reasonable assurance that (a) the aging effects will be managed, and (b) EQ components will continue to perform their intended function(s) consistent with the current licensing basis for the period of extended operation. Therefore, the DAEC EQ Program is an acceptable aging management program for license renewal under 10 CFR 54.21(c)(1)(iii) during the period of extended operation.

18.3.4 FATIGUE OF PRIMARY CONTAINMENT, PIPING, AND COMPONENTS

The Mark I analyses are detailed in the DAEC Plant Unique Analysis Report (PUAR) and assume 60 multiple SRV lifts and 740 single SRV lifts. Since these analyses include fatigue evaluations based on the occurrence of a limited number of transient cycles during the current licensed term of operation (40 years), they are TLAAs.

18.3.4.1 FATIGUE ANALYSIS OF SUPPRESSION CHAMBER

The maximum CUF (for 40 years) for the torus shell and welds is 0.467. Multiplying this value by 60/40 results in a CUF (for 60 years) of 0.70, which is less than 1.0.

2013-015 Since the 60 year CUF is less than 1.0, the current calculation remains valid for the period of extended operation.

18.3.4.2 FATIGUE ANALYSIS OF THE VENT SYSTEM AND VENT LINE BELLOWS

The maximum CUF (for 40 years) for the vent system components and welds is 0.33. Multiplying this value by 60/40 results in a CUF (for 60 years) of 0.50, which is less than 1.0.

For the vent line bellows, multiplying the number of thermal load cycles by 60/40 results in 225 cycles, which remains below rated capacity.

18.3.4.3 FATIGUE ANALYSIS OF SUPPRESSION CHAMBER EXTERNAL PIPING AND PENETRATIONS

The Mark I Owners Group prepared and submitted a generic fatigue evaluation report which addressed fatigue on a generic basis, and reported cumulative usage factors below 0.5. Conservatively multiplying this value by 1.5 (60 years divided by 40 years) results in 60-year CUFs of 0.75, which are less than 1.0 and therefore, acceptable.

The Mark I analyses assume 740 single SRV lifts and 60 multiple SRV lifts. A projection of the number of SRV lifts results in a projection of 334 single SRV lifts and 42 multiple lifts for 60 years. Both of these numbers are well below the values assumed in the Mark I analyses. Therefore the analyses remain valid for 60 years.

18.3.4.4 STRESS REPORT - CONTAINMENT VESSEL DESIGN CALCULATIONS

The Containment Vessel Stress Report includes a fatigue analysis exemption which is based on an assumed number of cycles. After increasing this number of cycles (for 60 years), the containment vessel remains exempt from fatigue analysis.

18.3.4.5 DESIGN ANALYSES OF FLUED HEADS FOR CLASS 1 PENETRATIONS

The analyses include the verification of adequacy of the flued heads by comparing allowable stresses (based on an assumed number of cycles) and maximum stress intensities. The stress results remain acceptable for the increased number of cycles for 60 years.

18.3.5 OTHER PLANT-SPECIFIC TLAAS

18.3.5.1 CRANES - REACTOR AND TURBINE BUILDING

The Turbine and Reactor Building Crane specifications assume a 40-year useful life for fatigue stress analysis purposes. The Ederer Generic Licensing Topical Report EDR-1(NP)–A, applicable to the Reactor Building Crane describes the crane as CMAA Class A crane. There are implicit cycle requirements for cranes designed in accordance with Crane Manufacturers Association of America (CMAA)-70. The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

18.3.5.2 EVALUATION OF THE FATIGUE LIFE OF THE STABILIZER ASSEMBLY

The evaluation of the fatigue life of the stabilizer assembly between the bioshield wall and the containment determined that the system is qualified for at least 400 fatigue cycles. The number of cycles expected for a 60-year life remains below 400.

18.3.5.3 EVALUATION OF EXISTING HCC-B002 “DOLLAR WELD” INDICATION

During RFO 17, inspections identified an indication in a circumferential weld in the reactor head (Vessel Head Dollar Weld HCC-B002) that did not meet ASME Section Xl IWB-3500 acceptance standards. The indication was evaluated and determined to be acceptable to leave as-is (IWB-3600 evaluation).

Per BWRVIP-74-A, a re-evaluation shall be performed for the 60 year service period corresponding to the LR term. The re-evaluation determined that the maximum end-of-service life (54 EFPY) applied stress intensity factor calculated for the indication is well below the material fracture toughness. Therefore, the existing flaw is acceptable for 60 year life.

18.3.5.4 EVALUATION OF THERMAL FATIGUE EFFECTS ON STEAM LEAD AND INLET TO RPV

This calculation evaluates the potential thermal fatigue effects on the steam lead into the condensing pot and the inlet to the RPV. The calculation was re-evaluated for the additional cycles that would be incurred during a 60 year life, with acceptable results.

18.3.5.5 CONTROL ROD DRIVE MECHANISM FATIGUE

The analysis for cyclic operation of the Control Rod Drive Mechanisms (CRDMs) resulted in a maximum cumulative usage factor (CUF) of 0.15 for the limiting CRD main flange at EPU conditions. Increasing this CUF by using a 1.5 multiplier (60 years/40 years) results in a 60-year CUF less than 1.0, which is acceptable.

The CUFs for the insert/withdrawal lines, discharge piping, scram monitoring stations and scram headers remain below 1.0 for 60 years, and are therefore acceptable.

18.3.5.6 MAIN STEAM ISOLATION VALVE D EVALUATION

A flaw evaluation was performed for the subsurface indications identified in the body of the D outboard steam isolation valve per ASME IWB-3600.

The assumed 40 years of operation from the last radioagraphy, bounding flaw sizes evaluated, margin to acceptance criteria, and commitment to perform another radiographic inspection of the repair when the valve is disassembled for other reasons provide adequate basis for the 60 year service period.

18.3.5.7 BELLOWS DESIGN ANALYSIS

The design analyses for Reactor Water Cleanup Supply penetration X-15 and RPV Feedwater penetration X-9A/B include cycle assumptions. Multiplying the design cycles by 1.5 (60 years/40 years), it is seen that the numbers of design cycles remain below the numbers of rated cycles; the results therefore remain acceptable.

18.4 LIST OF LICENSE RENEWAL COMMITMENTS

Duane Arnold License Renewal Commitments are located in UFSAR Table 18-1.

18.5 REFERENCES

1. 10 CFR 54.21(d) – Contents of Application – Technical Information

2. NUREG-1800 – Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants

3. NUREG-1801 – Generic Aging Lessons Learned (GALL) Report

2013-007 4. Letter, R. Anderson (NextEra Energy) to NRC, "License Renewal Commitment Changes," NG-13-0274, dated September 6, 2013.

TABLE 18-1
DUANE ARNOLD LICENSE RENEWAL COMMITMENTS

Item No.	System, Component or Program	Commitment[1]	Section	Schedule
1.	Buried Piping and Tanks Inspection Program	Implement Buried Piping and Tank Program	18.1.7	Prior to the period of extended operation
2.	BWR Vessel Internals Program	Perform an EVT-1 inspection of 5% of the top guide locations	18.1.14	Within six years of entering the period of extended operation
3.	BWR Vessel Internals Program	Perform an EVT-1 inspection of an additional 5% of the top guide locations	18.1.14	Within 12 years of entering the period of extended operation

[1] In the preceding table, the term "implement" means that the program is described in an approved procedure or other approved formal document; the test, inspection or monitoring procedure has been developed and approved; and the first test, inspection or monitoring activity has been scheduled.

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Item No.	System, Component or Program	Commitment[1]	Section	Schedule
4.	Electrical Cables and Connections Program	Implement an Electrical Cables and Connections Program and complete the first inspection prior to the period of extended operation.	18.1.17	Prior to the period of extended operation
5.	Electrical Cables and Connections Used in Instrumentation Circuits Program	Implement an Electrical Cables and Connections Used in Instrumentation Circuits Program and complete the first inspection prior to the period of extended operation.	18.1.18	Prior to the period of extended operation
6.	Electrical Connections Program	Implement an Electrical Connections Program and complete the one time inspection prior to the period of extended operation.	18.1.19	Prior to the period of extended operation
7.	Electrical Penetration Assemblies Program	Implement an Electrical Penetration Assemblies Program.	18.1.20	Prior to the period of extended operation
8.	External Surfaces Monitoring Program	Revise the inspection program to address inspector qualifications, types of components, degradation mechanisms, aging effects, acceptance criteria, inspection frequency, and periodic reviews to determine program effectiveness. The program will also specifically address inaccessible areas and include inspections of opportunity for possible corrosion under insulation.	18.1.21	Prior to the period of extended operation

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Item No.	System, Component or Program	Commitment[1]	Section	Schedule
9.	Fire Protection Program	The DAEC Fire Barrier Penetration Seal Inspection surveillance procedure will be enhanced to include criteria for visual inspections of fire barrier wall, ceiling and floors to examine for any sign of degradation such as cracking, spalling and loss of material caused by freeze-thaw, chemical attack and reaction with aggregates by fire protection qualified inspectors.	18.1.22	Prior to the period of extended operation
10.	Fire Protection Program	Enhance procedures to inspect the entire diesel driven fire pump fuel supply line for age related degradation.	18.1.22	Prior to the period of extended operation
11.	Fire Water System Program	Implement maintenance activities to perform volumetric examinations for pipe wall thinning of fire protection piping periodically during the period of extended operation.	18.1.23	Prior to the period of extended operation
12.	Fire Water System Program	Enhance procedures to include NFPA 25 criteria for sprinklers regarding replacing or testing	18.1.23	Prior to the period of extended operation
13.	Fire Water System Program	Enhance procedures to perform visual inspection of fire hydrants annually	18.1.23	Prior to the period of extended operation

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Item No.	System, Component or Program	Commitment[1]	Section	Schedule
14.	Fuel Oil Chemistry Program	Revise the program to require particulate testing of fuel oil samples from the diesel fire pump day tank	18.1.25	Prior to the period of extended operation
15.	Fuel Oil Chemistry Program	Enhance procedures to require sampling and testing of new fuel oil delivered to the diesel fire pump day tank; and to require that purchase orders and sampling procedures for diesel fuel delivered to and stored in the diesel fire pump day tank prohibit the delivery and use of biodiesel fuel.	18.1.25	Prior to the period of extended operation
16.	Fuel Oil Chemistry Program	Enhance procedures to perform periodic (10 year) draining, cleaning and visual inspection of the diesel fuel oil day tanks, diesel fire pump day tank, and diesel driven air start air compressor fuel oil tanks.	18.1.25	Prior to the period of extended operation
17.	Fuel Oil Chemistry Program	Implement procedures to require bottom thickness testing of the Standby Diesel Generator Day Tanks and the Diesel Fire Pump Day Tank.	18.1.25	Prior to the period of extended operation
2013-015 18.	Fuse Holders Program	Implement a Fuse Holders Program and complete the first test prior to the period of extended operation.	18.1.26	Prior to the period of extended operation

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Item No.	System, Component or Program	Commitment[1]	Section	Schedule
19.	Inaccessible Medium Voltage Cable Program	Implement an Inaccessible Medium Voltage Cable Program and complete the first inspection or test prior to the period of extended operation.	18.1.27	Prior to the period of extended operation
20.	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program	Implement an Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program.	18.1.28	Prior to the period of extended operation
21.	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems Program	Enhance procedures to monitor for corrosion and wear of the supporting steel and rails	18.1.29	Prior to the period of extended operation
2013-017 22.	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems Program	Enhance procedures to record over-capacity lifts for the reactor building and turbine building cranes	18.1.29	Prior to the period of extended operation

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Item No.	System, Component or Program	Commitment[1]	Section	Schedule
23.	Lubricating Oil Analysis Program	Enhance procedures to include diesel fire pump	18.1.30	Prior to the period of extended operation
24.	Metal Enclosed Bus Program	Implement a Metal Enclosed Bus Program and complete the first inspection prior to the period of extended operation.	18.1.31	Prior to the extended operation
25.	One-Time Inspection Program	Implement a One-Time Inspection Program and complete the one-time inspections prior to the period of extended operation.	18.1.32	Prior to the period of extended operation
26.	Reactor Vessel Surveillance Program	Implement a procedure to evaluate the BWRVIP ISP data as it becomes available.	18.1.35	Prior to the period of extended operation
27.	Reactor Vessel Surveillance Program BWRVIP-74-A BWR RPV Inspection and Flaw Evaluation Guidelines for License Renewal	Revise the Reactor Vessel Surveillance Program to implement the recommendations of BWRVIP-116 BWR Vessel and Internals Project Integrated Surveillance Program Implementation for License Renewal.	18.1.35	Prior to the period of extended operation

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Item No.	System, Component or Program	Commitment[1]	Section	Schedule
28.	Reactor Vessel Surveillance Program	Implement BWRVIP-116 with the conditions documented in Sections 3 and 4 of the NRC Staff's SE dated March 1, 2006 for BWRVIP-116, including the following: • NRC approval will be obtained for any change in the withdrawal schedules of the DAEC Reactor Vessel surveillance capsules. • If a standby capsule is removed from the DAEC Reactor Vessel without the intent to test it, the capsule will be stored in a manner which maintains it in a condition which would permit its future use, including during the period of extended operation, if necessary.	18.1.35	Prior to the period of extended operation
29.	Selective Leaching of Materials Program	Implement and complete a program to include one-time visual inspection and hardness measurement of selected components susceptible to selective leaching.	18.1.36	Prior to the period of extended operation
30.	Structures Monitoring Program	Enhance procedures to include structures and structural components not currently in Maintenance Rule Program	18.1.37	Prior to the period of extended operation

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31.	Structures Monitoring Program	Enhance procedures to include periodic sampling of groundwater for pH, chloride and sulfate concentration on a 5 year periodicity.	18.1.37	Prior to entering the period of extended operation
32.	Structures Monitoring Program	Enhance procedures to include a elastomer inspection to prevent leakage through containment penetration	18.1.37	Prior to the period of extended operation
33.	Structures Monitoring Program	Enhance procedures to include a requirement to contact the proper personnel to allow opportunistic inspection of the buried concrete foundation	18.1.37	Prior to the period of extended operation
34.	Structures Monitoring Program	Enhance procedures to include opportunistic inspections of the buried concrete foundation on a 10 year periodicity	18.1.37	Prior to the period of extended operation
35.	Metal Fatigue of Reactor Vessel Coolant Pressure Boundary Program	Enhance procedures to incorporate the requirements of NUREG/CR-6260 locations into the implementing procedures	18.2.2	Prior to the period of extended operation

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2017-014

36.	Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program	Enhance the Augmented Inspection procedures for the BWR Vessel Internals Program implementing procedures to include assessment for thermal and/or neutron embrittlement in susceptible Cast Austenitic Stainless Steel (CASS) components.	18.1.38	Prior to the period of extended operation
37.	Reactor Internals	DAEC will ensure that aging of core plate hold down bolts is appropriately addressed by completing one of the following actions: ▪ Install core plate wedges to eliminate the function of core plate hold down bolts. ▪ Perform analysis of the core plate rim hold down bolts that demonstrates adequacy to perform their intended function including loss of pre-load in the period of extended operation including the effects of projected neutron fluence. Inspection of core plate hold down bolts will be performed in accordance with BWRVIP-25, or a deviation disposition will be developed/submitted in accordance with BWRVIP-94.	18.1.14 18.3.1.7	Prior to entering the period of extended operation
38.	Reactor Vessel Circumferential Weld TLAA	Submit a relief request to address the frequency requirements of the inservice inspection of the RPV circumferential welds. (BWRVIP-05)	18.3.1.4	Prior to the period of extended operation

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39.	Quality Assurance Program (Corrective Action, Confirmation Process, Administrative Controls)	Expand the scope of its 10 CFR Part 50, Appendix B Quality Assurance program to include non-safety-related structures and components subject to an AMR for license renewal.	UFSAR 17.1.2	Prior to the period of extended operation
40.	Operating Experience	Perform an operating experience review of extended power uprate and its impact on aging management programs for systems, structures, and components (SSCs) before entering the period of extended operation.		Prior to the period of extended operation
41.	Bolting Integrity Program	Revise the implementing procedures for the ASME Section XI Inservice Inspection Subsections IWB, IWC, and IWD Program; ASME Section XI Inservice Inspection, Subsection IWF Program; External Surfaces Monitoring Program, Structural Monitoring Program and Buried Piping and Tanks Program such that they specifically address the inspection of fasteners (bolting, washers, nuts, etc.) for signs of leakage, corrosion/loss of material, cracking, and loss of preload/loss of prestress, as applicable.	18.1.6	Prior to the period of extended operation

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42.	BWR Penetrations Program	The implementing document for the BWR Penetrations Program will be revised to specify that guidance in BWRVIP-14, -59 and -60 will be used, as appropriate, depending on material, in the evaluation of crack growth in stainless steel, nickel alloys and low-alloy steels, respectively, when flaws are identified and evaluation required.	18.1.10	Prior to the period of extended operation
43.	Fire Protection Program	The DAEC Fire Barrier Penetration Seal Inspection surveillance procedure will be enhanced to ensure a approximately 10% of each type of penetration seal is included in the 35 percent selection of fire penetration seals that are visually inspected at an 18 month interval.	18.1.22	Prior to the period of extended operation
44.	Fire Protection Program	The DAEC Surveillance Procedure for the CO2 Cardox System Operability Annual Test will be enhanced to include a step to perform an inspection for corrosion and mechanical damage to system components.	18.1.22	Prior to the period of extended operation

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45.	ASME Class 1 Small-bore Piping Inspection Program	Implement an ASME Code Class 1 Small-bore Piping Inspection Program DAEC will perform volumetric examination of a minimum of ten percent of the ASME Code Class 1 small-bore socket welds each inspection interval. The ASME Code Class 1 Small-bore Piping inspection program will include provisions that a destructive examination may be performed on an opportunistic basis in lieu of the socket weld volumetric examinations.	18.1.40	Prior to the period of extended operation
46.	BWR Vessel Internals Program	The BWR Vessel Internals Program will incorporate the crack growth rate evaluations specified in the BWRVIP-100-A report. Plant specific inspection intervals will be developed for DAEC core shroud welds that are exposed to a neutron fluence value equal to or greater than 1 X 10^{21} n/cm^2 (E > 1 MeV), as needed.	18.1.14	Prior to the period of extended operation
47.		Not Used		
48.	Boral Surveillance Program	Implement a Boral Surveillance Program and complete the first in-situ neutron attenuation test of the PaR spent fuel racks.	18.1.41	Prior to the period of extended operation
49.	Fire Protection Program	Enhance procedures to inspect the 1 hour fire rated gypsum board wall that separates the control room computer room area from the front panel area for aging due to cracking.	18.1.22	Prior to the period of extended operation

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50.	ASME Section XI, Inservice Inspection, Subsection IWE Program	Perform recoating of suppression pool interior surfaces below the water line.	18.1.4	Complete recoating prior to startup from the first refuel outage during the period of extended operation.
51.	Metal Fatigue of Reactor Vessel Coolant Pressure Boundary Program	Future revisions/updates to the environmental fatigue calculations for the Recirculation Inlet Nozzle Safe End, Feedwater Nozzle Safe End, and Core Spray Nozzle Safe End will use F_{en} data for Nickel Alloy from the methodology that is described in NUREG/CR-6909 in the determination of usage factors.	18.2.2	Upon calculation revision.

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52.	Buried Piping and Tanks Inspection Program	Enhance the Buried Piping and Tanks Inspection Program to include inspection of at least a minimum number of pipe segments in each material group (one stainless steel, two carbon steel, one cast iron, and two ductile iron) prior to entry into the period of extended operation and each ten year period after entry into the period of extended operation. Where torsional guided wave data indicates significant susceptibility, inspections will be performed on associated locations. The sample locations for directed inspections will preferentially select higher risk locations. Piping that normally contains hazardous materials will be prioritized in the inspection location selection process. The diesel fuel oil piping will be inspected prior to entry into the period of extended operation. These directed inspections will be performed with sufficient excavation to expose at least ten linear feet of piping as practicable, including the pipe bottom. Inspections of coated carbon steel piping will include the coating and backfill in the vicinity of the piping for material that could cause coating damage. The uncoated stainless steel, ductile iron and cast iron piping will be externally inspected for corrosion, and the fill in the vicinity of the piping will be inspected for material that could cause external damage to the stainless steel, ductile iron or cast iron pipe. (2013-007) As alternatives to direct external inspections involving excavation, the buried piping inspections may be performed by a hydrostatic test on at least 25% of the code class/safety-related or hazmat piping or both constructed from the material under consideration on an interval not to exceed 5 years. Or, similarly, at least 25% of the code class/safety-related or hazmat piping or both constructed from the material under consideration will be internally inspected by a method capable of precisely determining pipe wall thickness on an interval not to exceed 5 years.	18.1.7	Prior to the period of extended operation

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53.	Buried Piping and Tanks Inspection Program	Cathodic protection system availability will be maintained $\geq$ 90%. If 90% availability is not maintained, the condition will be entered into the corrective action program to evaluate the impact and take corrective actions. Availabilty will be demonstrated by having no more than six months of rectifier out-of-service time in any sixty month period, as determined on a "per rectifier" basis; or no more than six months of rectifier out-of-service time in any twelve month period, for all rectifiers combined. Annual surveys will continue to be performed in accordance with NACE Standard Practice.	18.1.7	Prior to the period of extended operation
54.	Inaccessible Medium Voltage Cable Program	The program will be enhanced to include 480 V to 2 kV power cables. This includes enhancing the scope of the program, preventive actions, parameters monitored or inspected, detection of aging effects and operating experience.	18.1.27	Prior to the period of extended operation

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<table>
<tr><td rowspan="5">55.</td><td rowspan="5">One-Time Inspection Program</td><td colspan="3">The sample selection for the DAEC One-Time Inspection program will include a representative sample of the population. Existing maintenance records that document component condition will be used as part of the sample. The material environment combinations and the number of required inspections for the sample group will be as shown in the following table:</td><td rowspan="5">18.1.32</td><td rowspan="5">Prior to the period of extended operation</td></tr>
<tr><td>Sample Group Environment</td><td>Materials in the Sample Group Environment</td><td>Number of Required Inspections</td></tr>
<tr><td>Sample Group 1 – Fuel Oil</td><td>Carbon Steel and Cast Iron
Stainless Steel</td><td>6</td></tr>
<tr><td>Sample Group 2 – Lube Oil</td><td>Aluminum alloy
Carbon Steel and Cast Iron
Copper Alloy including Admiralty Brass
Stainless Steel including Cast Austenitic Stainless Steel</td><td>38</td></tr>
<tr><td>Sample Group 3/4-Steam and Treated Water, Reactor Coolant and Sodium Pentaborate</td><td>Copper Alloy, Copper, Admiralty Brass, Brass, Carbon Steel, Low Alloy Steel and Cast Iron Stainless Steel Including Cast Austenitic Stainless Steel, Stainless Steel Cladding of Low Alloy Steel with Carbon Steel and Nickel Alloy</td><td>57</td></tr>
</table>

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56.	Selective Leaching Program	The sample selection for the DAEC Selective Leaching program will include a representative sample of approximately 20% of the population for each susceptible material group or a maximum of 25 components. Existing maintenance records that document component condition will be used as part of the sample.	18.1.36	Prior to the period of extended operation
57.	Structures Monitoring Program	The DAEC Structures Monitoring Program will be enhanced to incorporate quantitative acceptance criteria for concrete inspections of all in-scope structures as determined from reviewing ACI 349.3R-96. Enhancements will be made to the program prior to entry into the period of extended operation. Conditions that are acceptable without further evaluation (ACI 349.3R-96 Section 5.1) observed during visual surveys will not be documented in the survey reports if the inspection is performed by a “responsible engineer” as defined in ACI 349.3R-96 Section 7.	18.1.37	Prior to the period of extended operation

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58.	TLAA	DAEC will perform a review of usage factors for ASME Class 1 components with design basis calculations to determine whether the NUREG/CR-6260-based components that have been evaluated for the effects of the reactor coolant environment on fatigue usage are the limiting components for the DAEC plant configuration. This review includes qualitative or quantitative comparisons of components. If more limiting components are identified, the most limiting component usage factor will be evaluated for the effects of the reactor cooling environment on fatigue usage prior to entry into the period of extended operation. If a new limiting component identified consist of nickel alloy, the methodology used to perform environmentally-assisted fatigue calculation for nickel alloy will be consistent with NUREG/CR-6909.	18.3.2.4	Prior to the period of extended operation