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QUAD CITIES, UNITS 1 AND 2
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Introduction

The summary descriptions of aging management program activities presented in this Appendix A represent commitments for managing aging of the systems, structures and components within the scope of license renewal during the period of extended operation. This appendix also provides summary descriptions of time-limited aging analyses. The aging management programs described below have been implemented.

A.1 AGING MANAGEMENT PROGRAMS

A.1.1 ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD

The ASME Section XI Inservice Inspection, Subsections IWB, IWC and IWD aging management program consists of periodic volumetric and visual examinations of components for assessment, identification of signs of degradation, and establishment of corrective actions. The inspections have been implemented in accordance with 10 CFR 50.55a.

Quad Cities has implemented the guidance of BWRVIP-74, "BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines," with the following exception. Exception: Risk Informed Inservice Inspection is implemented in lieu of ASME Section XI requirements for portions of Class 1 and Class 2 systems.

Technical Specification revisions containing new P-T Curves were submitted prior to the term of extended operation.

A.1.2 Water Chemistry

The water chemistry aging management program consists of monitoring and control of water chemistry to keep peak levels of various contaminants below system-specific limits based on industry-recognized guidelines of EPRI 3002002623, "BWR Water Chemistry Guidelines." To mitigate aging effects on component surfaces that are exposed to water as process fluid, the chemistry programs are used to control water chemistry for impurities (e.g., chlorides, and sulfates) that accelerate corrosion.

Quad Cities has implemented the general guidance provided in BWRVIP 190 Revision 1, "BWR Water Chemistry Guidelines" (EPRI Report 3002002623).

A.1.3 Reactor Head Closure Studs

The reactor head closure studs aging management program includes inservice inspection (ISI). This program also includes preventive actions and inspection techniques for BWRs. The requirements of ASME Section XI have been implemented in accordance with 10 CFR 50.55a. The reactor head studs are not metal-plated, and have had manganese phosphate coatings applied.

A.1.4 BWR Vessel ID Attachment Welds

The BWR vessel ID attachment welds aging management program includes (a) inspection and flaw evaluation in conformance with the guidelines of staff-approved Boiling Water Reactor Vessel and Internals Project BWRVIP-48, “Vessel ID Attachment Weld Inspection and Evaluation Guidelines,” and/or ASME Section XI; and (b) monitoring and control of reactor coolant water chemistry in accordance with industry-recognized guidelines of EPRI 3002002623, “BWR Water Chemistry Guidelines.” The requirements of ASME Section XI have been implemented in accordance with 10 CFR 50.55a.

A.1.5 BWR Feedwater Nozzle

The BWR feedwater nozzle aging management program includes enhancing the inservice inspections (ISI) specified in the ASME Code, Section XI, with the recommendation of General Electric (GE) NE-523-A71-0594-A, Revision 01, “Alternate BWR Feedwater Nozzle Inspection Requirements," to perform periodic ultrasonic testing inspection of critical regions of the BWR feedwater nozzles.

A.1.6 BWR Control Rod Drive Return Line Nozzle

The BWR control rod drive return line nozzle aging management program consists of previously implemented system modifications and inservice inspections that manage the aging effect of cracking in the control rod drive return line nozzles. The control rod drive return line nozzles have been capped. Inservice inspections are performed consistent with ASME Section XI requirements. No augmented inspections in accordance with NUREG-0619, “BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking,” or the alternative recommendations of GE NE-523-A71-0594, “Alternate BWR Feedwater Nozzle Inspection Requirements," are required. The requirements of ASME Section XI have been implemented in accordance with 10 CFR 50.55a.

A.1.7 BWR Stress Corrosion Cracking

The BWR stress corrosion cracking aging management program to manage intergranular stress corrosion cracking (IGSCC) in boiling water reactor coolant pressure boundary piping four inches and larger nominal pipe size made of stainless steel (SS) is delineated, in part, in NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," Revision 2, BWRVIP 75, “Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules,” and Nuclear Regulatory Commission (NRC) Generic Letter (GL) 88-01, “NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping", and its Supplement 1. The program includes (a) replacements and preventive measures to mitigate IGSCC and (b) inspections to monitor IGSCC and its effects. Water chemistry is monitored and maintained in accordance with industry-recognized guidelines in EPRI 3002002623, “BWR Water Chemistry Guidelines.” The requirements of ASME Section XI have been implemented in accordance with 10 CFR 50.55a.

Quad Cities has implemented the general guidance provided in BWRVIP-75, "Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules," with Exception – The Relief Request submitted for the implementation of RISI indicates the Category A Welds are "subsumed into the RISI program."

A.1.8 BWR Penetrations

The BWR penetrations aging management program includes (a) inspection and flaw evaluation in conformance with the guidelines of staff-approved Boiling Water Reactor Vessel and Internals Project (BWRVIP)-49, "Instrument Penetration Inspection and Flaw Evaluation Guidelines," and BWRVIP-27, "BWR Standby Liquid Control System/Core Plate Delta-P Inspection and Flaw Evaluation Guidelines," documents and (b) monitoring and control of reactor coolant water chemistry in accordance with industry-recognized guidelines of EPRI 3002002623, "BWR Water Chemistry Guidelines," to ensure the long-term integrity and safe operation of boiling water reactor vessel internal components. The requirements of ASME Section XI have been implemented in accordance with 10 CFR 50.55a.

A.1.9 BWR Vessel Internals

The BWR vessel internals aging management program includes (a) inspection and flaw evaluation in conformance with the guidelines of applicable and staff-approved Boiling Water Reactor Vessel and Internals Project (BWRVIP) documents, and with ASME Section XI; and (b) monitoring and control of reactor coolant water chemistry in accordance with industry-recognized guidelines of EPRI 3002002623, "BWR Water Chemistry Guidelines," to ensure the long-term integrity and safe operation of boiling water reactor vessel internal components. The requirements of ASME Section XI have been implemented in accordance with 10 CFR 50.55a.

Quad Cities has implemented the general guidance provided in BWRVIP-18, "BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines."

Quad Cities has implemented the general guidance provided in BWRVIP-25, "BWR Core Plate Inspection and Flaw Evaluation Guidelines."

Quad Cities has implemented the guidance provided in BWRVIP-26, "BWR Top Guide Inspection and Flaw Evaluation Guidelines." Additionally, Quad Cities will perform augmented inspections for the top guide similar to the inspections of control rod drive housing (CRDH) guide tubes.

Quad Cities has implemented the general guidance provided in BWRVIP-38, "BWR Shroud Support Inspection an Flaw Evaluation Guidelines." Quad Cities will perform the additional inspections of the lower plenum (i.e. shroud support leg welds) when new inspection techniques and tooling are developed, incorporated into the applicable BWRVIP document(s).

Quad Cities has implemented the general guidance provided in BWRVIP-41, "BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines." Quad Cities will perform the additional inspections of the inaccessible thermal sleeve welds when new inspection techniques and tooling are developed, incorporated into the applicable BWRVIP document(s).

Quad Cities has implemented the general guidance provided in BWRVIP-47, “BWR Lower Plenum Inspection and Flaw Evaluation Guidelines.”

Quad Cities has implemented the general guidance provided in BWRVIP-76, “BWR Core Shroud Inspection and Flaw Evaluation Guidelines.”

A.1.10 Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)

The thermal aging and neutron irradiation embrittlement of cast austenitic stainless steel (CASS) aging management program consists of (1) determination of the susceptibility of cast austenitic stainless steel components to thermal aging embrittlement, (2) accounting for the synergistic effects of thermal aging and neutron irradiation, and (3) implementing a supplemental examination program, as necessary. The program was implemented prior to the period of extended operation.

Quad Cities has implemented the general guidance provided in BWRVIP-234, "Thermal Aging and Neutron Embrittlement of Cast Authenitic Stainless Steels for BWR Internals."

Quad Cities has implemented the general guidance provided in EPRI BWRVIP Letter 2017-030, "BWRVIP Assessment of BWR Orificed Fuel Support Aging Management."

A.1.11 Flow-Accelerated Corrosion

The flow-accelerated corrosion aging management program consists of (1) appropriate analysis and baseline inspections, (2) determination of the extent of thinning, and replacement or repair of components, and (3) follow-up inspections to confirm or quantify effects, and to take longer-term corrective actions. This program is in response to NRC Generic Letter 89-08, “Erosion/Corrosion-Induced Pipe Wall Thinning.” The program relies on implementation of the EPRI NSAC-202L, “Recommendations for an Effective Flow-Accelerated Corrosion Program” guidelines. Prior to the period of extended operation the program was revised to include main steam and reactor head vent piping within the scope of license renewal.

A.1.12 Bolting Integrity

This bolting integrity aging management program incorporates industry recommendations of EPRI NP-5769, “Degradation and Failure of Bolting in Nuclear Power Plants,” and includes periodic visual inspections for external surface degradation that may be caused by loss of material or cracking of the bolting, or by an adverse environment. Inspection of inservice inspection Class 1, 2, and 3 components is conducted in accordance with ASME Section XI. The requirements of ASME Section XI have been implemented in accordance with 10 CFR 50.55a. The program includes inspections of bolted joints of diesel generator system components and of components in locations containing high humidity or moisture. In addition, the program includes inspections of the reactor vessel-to-ring girder bolting.

Program activities address the guidance contained in EPRI TR-104213, “Bolted Joint Maintenance and Applications Guide,” but do not specifically identify its use. Non-safety component inspections rely on detection of visible leakage during preventive maintenance and routine observation. The program does not address structural and component support bolting with the exception of the reactor vessel-to-ring girder bolting. The aging management of all other structural bolting is covered by the structures monitoring program. Aging management of ASME Section XI Class 1, 2, and 3 and Class MC (excluding Class MC piping) support members, including mechanical connections, is covered by the “ASME Section XI, Subsection IWF” aging management program. Aging management of Class MC piping support members, including mechanical connections is covered by the “Structures Monitoring” aging management program.

A.1.13 Open-Cycle Cooling Water System

The open-cycle cooling water system aging management program includes (a) surveillance and control of biofouling, (b) tests to verify heat transfer, (c) a routine inspection and maintenance program, including system flushing and chemical treatment, (d) periodic inspections for leakage, loss of material, and blockage, (e) engineering evaluations and heat sink performance assessments, and (f) assessments of the overall heat sink program. These evaluations and assessments produced specific component and programmatic corrective actions. The program provides assurance that the open-cycle cooling water system is in compliance with General Design Criteria, and with quality assurance requirements, to ensure that the open-cycle cooling water system can be managed for an extended period of operation. This program is in response to and uses the test and inspection guidelines of NRC Generic Letter 89-13, “Service Water System Problems Affecting Safety-Related Equipment.” Prior to the period of extended operation, the scope of the program was increased to include inspection of additional heat exchangers and sub-components, external surfaces of various submerged pumps and piping, cooling water pump linings, and components in the pump vaults that have a high humidity or moisture environment.

A.1.14 Closed-Cycle Cooling Water System

The closed-cycle cooling water system aging management program relies on preventive measures to minimize corrosion by maintaining inhibitors and by performing non-chemistry monitoring consisting of inspection and nondestructive examinations (NDEs) based on industry-recognized guidelines of EPRI 1007820, “Closed Cooling Water Chemistry Guidelines,” for closed-cycle cooling water systems. Station maintenance inspections and NDE provide condition monitoring of heat exchangers exposed to closed-cycle cooling water environments. Prior to the period of extended operation, the program was enhanced to include procedure revisions that provide for monitoring of specific chemistry parameters in order to meet EPRI 1007820 guidance.

A.1.15 Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems

The inspection of overhead heavy load and light load (related to refueling) handling systems aging management program confirms the effectiveness of the maintenance monitoring program and the effects of past and future usage on the structural reliability of cranes and hoists. Administrative controls ensure that only allowable loads are handled, and fatigue failure of structural elements is not expected. A time-limited aging analysis concludes that there are no fatigue concerns for reactor building overhead cranes during the period of extended operation. The bridge, trolley, and other structural components are visually inspected on a routine basis for degradation. These cranes are included in the corporate structural monitoring program (which complies with the 10 CFR 50.65 maintenance rule) and in various station procedures. Prior to the period of extended operation, the program was enhanced to include inspections for rail wear and proper crane travel on rails, and corrosion of crane structural components.

A.1.16 Compressed Air Monitoring

The compressed air monitoring aging management program consists of inspection, monitoring, and testing of the entire system, including (1) pressure decay testing, visual inspections, and walkdowns of various system locations; and (2) preventive monitoring that checks air quality at various locations in the system to ensure that dewpoint, particulates, and suspended hydrocarbons are kept within the specified limits. This program is consistent with responses to NRC Generic Letter 88-14, “Instrument Air Supply Problems,” and ANSI/ISA-S7.3-1975, "Quality Standard for Instrument Air." Prior to the period of extended operation, the program was enhanced to include inspections of instrument air distribution piping based on EPRI TR-108147, “Compressor and Instrument Air System Maintenance Guide.”

A.1.17 BWR Reactor Water Cleanup System

The BWR reactor water cleanup (RWCU) system aging management program monitors and controls reactor water chemistry based on industry-recognized guidelines of EPRI 3002002623, “BWR Water Chemistry Guidelines," to reduce the susceptibility of RWCU piping to stress corrosion cracking (SCC) and intergranular stress corrosion cracking (IGSCC). RWCU system piping has been replaced with piping that is resistant to intergranular stress corrosion cracking, in response to NRC Generic Letter 88-01, “NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping," concerns. In addition, all actions requested in NRC Generic Letter 89-10, “Safety-Related Motor-Operated Valve Testing and Surveillance,” have been completed. Therefore, inservice inspection in accordance with ASME Section XI is not required.

A.1.18 Fire Protection

The fire protection aging management program includes a fire barrier inspection program and a diesel-driven fire pump inspection program. The fire barrier inspection program requires periodic visual inspection of fire barrier penetration seals; fire wraps and fire proofing; fire barrier walls, ceilings, and floors; flood barrier penetration seals that also serve as fire barrier seals; and periodic visual inspection and functional tests of fire rated doors to ensure that their operability is maintained. The program includes surveillance tests of fuel oil systems for the diesel-driven fire pumps to ensure that the fuel supply line can perform intended functions. The program also includes visual inspections and periodic operability tests of the carbon dioxide fire suppression system based on NFPA codes.

Prior to the period of extended operation, the program was revised to include:

- Inspection of external surfaces of the carbon dioxide systems
- Specific fuel supply leak inspection criteria for fire pumps
- Specific inspection criteria for fire doors

A.1.19 Fire Water System

The fire water system aging management program provides fire system header and hydrant flushing, system performance (flow and pressure) testing, and inspections, on a periodic basis; and for injection of chemical agents during or subsequent to flushing to minimize biofouling. System performance tests measure hydraulic resistance and compare results with previous testing. This approach eliminates the need for tests at maximum design flow and pressure. Internal inspections are conducted on system components when disassembled to identify evidence of corrosion or biofouling. Fire header pressure is maintained through a crosstie with the service water system. Significant leakage (exceeding the capacity of this line) would be identified by automatic start of the fire pumps, which would initiate immediate investigation and corrective action. Inspection and surveillance testing is performed in accordance with procedures based on applicable NFPA codes. Where code deviations are required or desirable, the intent of the code is maintained by technical justifications.

Sprinkler test requirements were modified prior to the period of extended operation to include sprinkler sampling in accordance with NFPA 25, “Inspection, Testing and Maintenance of Water-Based Fire Protection Systems,” Section 2-3.1. Samples will be submitted to a testing laboratory prior to being in service 50 years. This testing will be repeated at intervals not exceeding 10 years.

Prior to the period of extended operation the program was revised to include external surface inspections of submerged fire pumps, outdoor hydrants, and outdoor transformer deluge systems; and periodic non-intrusive wall thickness measurements of selected portions of the fire water system at intervals that do not exceed every 10 years.

A.1.20 Aboveground Carbon Steel Tanks

The aboveground carbon steel tanks aging management program manages corrosion of outdoor nitrogen tanks and aluminum storage tanks. Paint is a corrosion preventive measure, and periodic visual inspections monitor degradation of the paint and any resulting metal degradation. Carbon steel tanks in the scope of license renewal are above ground and not directly supported by earthen or concrete foundations. Therefore, inspection of the sealant or caulking at the tank-foundation interface, and inspection of inaccessible tank locations and on-grade tank bottoms do not apply.

Aluminum storage tanks within the scope of license renewal are supported by earthen/concrete foundations. The tank-foundation interfaces (including foundation coatings) are periodically inspected for degradation. Periodic visual inspections of the internal/external surfaces of the aluminum storage tanks are conducted.

Prior to the period of extended operation, the program was revised to include documentation of results of periodic system engineer walkdowns of the nitrogen tanks, periodic visual inspections of the internal/external surfaces of aluminum tanks, and a one-time internal ultrasonic inspection of the bottom of one aluminum storage tank.

A.1.21 Fuel Oil Chemistry

The fuel oil chemistry aging management program relies on a combination of surveillance and maintenance procedures. Monitoring and controlling fuel oil contamination maintains the fuel oil quality. Exposure to fuel oil contaminants such as water and microbiological organisms is minimized by routine draining and cleaning of fuel oil tanks, and by fuel oil sampling and analysis, including analysis of new oil before its introduction into the storage tanks. A biocide is added to the fuel oil storage tanks during each new fuel delivery. Sampling and testing of diesel fuel oil is in accordance with ASTM D2709, ASTM D4057 and ASTM D5452. Emergency diesel generator fuel oil analysis acceptance criteria are contained in the Technical Specifications and are based on the requirements of ASTM D975.

A.1.22 Reactor Vessel Surveillance

The reactor vessel surveillance aging management program includes periodic testing of metallurgical surveillance samples to monitor the progress of neutron embrittlement of the reactor pressure vessel as a function of neutron fluence, in accordance with Regulatory Guide (RG) 1.99, “Radiation Embrittlement of Reactor Vessel Materials," Revision 2.

As of September 2012, prior to and during the period of extended operation the program is consistent with NRC approved document BWRVIP-86, Revision 1-A, “Updated BWR Integrated Surveillance Program (ISP) Implementation Plan.”

The program will ensure coupon availability during the period of extended operation, and provide for saving withdrawn coupons for future reconstitution.

A.1.23 One-Time Inspection

The one-time inspection aging management program includes inspections of a number of samples of the piping and components listed below. The inspections were implemented prior to the period of extended operation to manage aging effects of selected components within the scope of license renewal. The purpose of the inspection was to determine if a specified aging effect is occurring. If the aging effect was occurring, an evaluation was performed to determine the effect it will have on the ability of affected components to perform their intended functions for the period of extended operation, and appropriate corrective action was taken. The program includes the following one-time inspections:

- Volumetric examination of 10% of the high and medium risk butt welds of Class I piping less than four inch nominal pipe size (NPS) exposed to reactor coolant for cracking.

- Inspection of a sample of torus saddle Lubrite baseplates for galvanic corrosion, wear, and lockup to confirm the condition of the inaccessible drywell radial beam Lubrite baseplates.

- Inspection of a sample of spent fuel pool cooling and demineralizer system components for corrosion in stagnant locations to verify effective water chemistry controls.

- Inspection of a sample of condensate and torus water components for corrosion and/or stress corrosion cracking in stagnant locations to verify effective water chemistry control.

- Inspection of a sample of compressed gas system piping components for corrosion, and a sample of compressed gas system flexible hoses for age-related degradation.

- Inspection of a sample of lower sections of carbon steel fuel oil and lubricating oil tanks for reduced thickness.

- Inspection of a sample of fuel oil and lubricating oil piping and components for corrosion.

- Inspection of a sample of standby gas treatment and ventilation system components for loss of material.

- Inspection of a sample of stainless steel standby liquid control (SBLC) system components not in the reactor coolant pressure boundary of the SBLC system for cracking, to verify effective water chemistry control.

- Inspection of a sample of HPCI turbine lubricating oil hoses for age-related degradation. Based on the completed Unit 1 inspection, these hoses are made of carbon steel, not elastomer as originally assumed. Therefore, no additional inspections are required.

- Inspection of a sample of non-safety related vents and drains including their valves and associated piping, for age-related degradation leading to a loss of structural integrity.

- Inspection of a sample of 10 CFR 54.4(a)(2) components for corrosion for which the component, material, environment, aging effect, or their combination is not specifically identified in NUREG-1801, "Generic Aging Lessons Learned (GALL) Report."

- Inspection of stainless steel clevis pins for crack initiation and growth due to SCC and the interfacing carbon steel support members for loss of material due to pitting and galvanic corrosion in torus water environment.

A.1.24 Selective Leaching of Materials

The selective leaching of materials aging management program includes numerous one-time inspections of components of the different susceptible materials to determine if loss of material due to selective leaching is occurring. The visual examinations will be conducted by personnel certified to perform VT-1 examinations and will be performed in accordance with approved work instructions. The work instructions will include guidance for identifying the aging management effect of concern. If selective leaching is occurring the program requires evaluation of the effect it will have on the ability of the affected components to perform their intended functions for the period of extended operation, and of the need to expand the test sample. For systems subjected to environments where water is not treated (i.e., the open-cycle cooling water system) the program also follows the guidance of NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment." NUREG-1801 indicates that the selective leaching of materials aging management program includes one-time hardness measurements of a selected set of components. Visual inspections supplemented by other examinations in lieu of hardness tests of the selected components will be performed.

A.1.25 Buried Piping and Tanks Inspection

The buried piping and tanks inspection aging management program includes (1) preventive measures to mitigate corrosion, and (2) periodic inspection to manage the effects of corrosion on the pressure-retaining capacity of buried carbon steel piping and tanks. The program includes the use of piping and component coatings and wrappings, periodic pressure testing, buried tank leakage checks, inspections of buried tank interior surfaces, one-time visual inspection of exterior surface of section of ductile iron fire protection piping, and inspections of the ground above buried tanks and piping.

Prior to the period of extended operation a one-time visual inspection of the external surface of a buried piping section, and a one-time internal ultrasonic inspection of a sampling of the buried steel tanks was performed.

A.1.26 ASME Section XI, Subsection IWE

The ASME Section XI, Subsection IWE aging management program consists of periodic visual examination for signs of degradation, and limited surface or volumetric examination when augmented examination is required. The program covers steel containment shells and their integral attachments; containment hatches and airlocks; seals, gaskets and moisture barriers; and pressure-retaining bolting. The program includes assessment of damage and corrective actions. The requirements of ASME Section XI have been implemented in accordance with 10 CFR 50.55a.

A.1.27 ASME Section XI, Subsection IWF

The ASME Section XI, Subsection IWF aging management program consists of periodic visual examination of ASME Section XI Class 1, 2, and 3 component and piping supports for signs of degradation, evaluation, and establishment of corrective actions. The requirements of ASME Section XI have been implemented in accordance with 10 CFR 50.55a. Prior to the period of extended operation the program was revised to include ASME Class MC non-piping component supports: Biological Shield to Containment Stabilizer, RPV Male Stabilizer Attached to Outside of Drywell Shell, RPV Female Stabilizer and Anchor Rods, Suppression Chamber Ring Girder Vertical Supports and Base Plates, Suppression Chamber Saddle Supports and Base Plates, Suppression Chamber Seismic Restraints and Base Plates, and Vent Header Vertical Column Supports.

A.1.28 10 CFR Part 50, Appendix J

The 10 CFR Part 50, Appendix J aging management program monitors leakage rates through the containment pressure boundary, including the drywell and torus, penetrations, fittings, and other access openings; in order to detect degradation of containment pressure boundary. Corrective actions are taken if leakage rates exceed acceptance criteria. The Appendix J program also manages changes in material properties of gaskets, o-rings, and packing materials for the containment pressure boundary access points. The containment leak rate tests are performed in accordance with the regulations and guidance provided in 10 CFR 50 Appendix J Option B, Regulatory Guide 1.163, “Performance-Based Containment Leak-Testing Program," NEI 94-01, “Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50 Appendix J,” and ANSI/ANS 56.8, “Containment System Leakage Testing Requirements,” and an alternate test that was accepted by NRC granting the exemption for primary containment penetration expansion bellows assemblies that use a two ply design.

A.1.29 Masonry Wall Program

This masonry wall aging management program consists of inspections, based on IE Bulletin 80-11, "Masonry Wall Design," and plant-specific monitoring proposed by IN 87-67, "Lessons Learned from Regional Inspections of Licensee Actions in Response to IE Bulletin 80-11," for managing cracking of masonry walls. This program is part of the structures monitoring program.

A.1.30 Structures Monitoring Program

The structures monitoring aging management program includes periodic inspection and monitoring of the condition of structures; supports not included in the “ASME Section XI, Subsection IWF” aging management program; and external surfaces of mechanical and electrical components. The program ensures that aging degradation leading to loss of intended functions will be detected and that the extent of degradation can be determined. This program was developed under 10 CFR 50.65 and is based on NUMARC 93-01, “Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” Revision 4A and Regulatory Guide 1.160, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” Revision 3.

Prior to the period of extended operation the program was revised to include:

- Inspections of structural steel components in secondary containment, flood barriers, electrical panels and racks, junction boxes, instrument panels and racks, offsite power structural components and their foundations.
- Periodic reviews of chemistry data on below-grade water to confirm that the environment remains non-aggressive for aggressive chemical attack of concrete or corrosion of embedded steel.
- Inspection of a sample of non-insulated indoor piping external surfaces at locations immediately adjacent to periodically inspected piping supports.
- Reference to specific insulation inspection criteria for existing cold weather preparation and inspection procedures for outdoor insulation, and the establishment of new inspections for various indoor area piping and equipment insulation.
- Addition of specific inspection parameters for non-structural joints, roofing, grout pads and isolation gaps.
- Extension of inspection criteria to the structural steel, concrete, masonry walls, equipment foundations, and component support sections of the program.
- Addition of inspection criteria for standard components such as snubbers, struts, and spring cans.
- A VT-3 visual inspection of 15% of the non-exempt Class MC pipe supports once every 10 years.
- Management of the aging effects of those major components and larger piping headers that were credited as an anchor for non-safety related piping.

A.1.31 RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants

The RG 1.127, “Inspection of Water-Control Structures Associated with Nuclear Power Plants,” aging management program consists of inspection and surveillance of structural steel elements (exposed to raw water) and concrete (exposed and not exposed to raw water) that are in the crib house and discharge canal weir structure supporting the ultimate heat sink and within the scope of license renewal and the earthen walls of the intake and discharge flumes/canals. The activities are based on Regulatory Guide 1.127, Revision 1, and are part of the structures monitoring program. Prior to the period of extended operation the program was revised to include monitoring crib house concrete walls and slabs with opposing sides in contact with river water, and the discharge canal weir supporting the ultimate heat sink; to emphasize inspection for structural integrity of concrete and steel components; and to identify specific types of components to be inspected.

A.1.32 Protective Coating Monitoring and Maintenance Program

The protective coating monitoring and maintenance aging management program consists of guidance for selection, application, inspection, and maintenance of Service Level I protective coatings. This program is implemented in accordance with Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants," Revision 0, ANSI N101 4-1972, "Quality Assurance for Protective Coatings Applied to Nuclear Facilities," and the guidance of EPRI TR-109937, "Guidelines on Nuclear Safety-Related Coating." Prior to the period of extended operation the program was revised to include thorough visual inspection of Service Level 1 coatings near sumps or screens for the emergency core cooling system, pre-inspection review of previous reports so that trends can be identified, and analysis of suspected causes of any coating failures.

A.1.33 Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

The electrical cables and connections not subject to 10 CFR 50.49 environmental qualification requirements aging management program manages aging of cables and connections which might be susceptible to aging during the period of extended operation. All accessible electrical cables and connections installed in adverse localized environments are visually inspected at least once every 10 years for indications of accelerated insulation aging. An adverse localized environment is a condition in a limited plant area that is significantly more severe than the specified service environment for a subject electrical cable or connection. This is a new program initiated prior to the period of extended operation.

A.1.34 Metal Fatigue of Reactor Coolant Pressure Boundary

The metal fatigue and reactor coolant pressure boundary aging management program ensures that the design fatigue usage factor limit will not be exceeded during the period of extended operation. The program was enhanced prior to the period of extended operation. The enhanced program calculates and tracks cumulative usage factors for bounding locations in the reactor coolant pressure boundary (reactor pressure vessel and Class I piping), containment torus, torus vents, and torus attached piping and penetrations. The enhanced program uses the EPRI-licensed FatiguePro® cycle counting and fatigue usage factor tracking computer program, which provides for calculation of stress cycles and fatigue usage factors from operating cycles, automated counting of fatigue stress cycles, and automated calculation and tracking of fatigue cumulative usage factors.

A.1.35 Environmental Qualification (EQ) of Electrical Components

The effects of aging on the intended functions will be adequately managed per the requirements of 10 CFR 54.21 (c)(1)(iii). The existing environmental qualification (EQ) program manages aging of electrical equipment within the scope of 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," for the period of extended operation. The program establishes, demonstrates, and documents the level of qualification, qualified configurations, maintenance, surveillance and replacements necessary to meet 10 CFR 50.49. A qualified life is determined for equipment within the scope of the program and appropriate actions such as replacement or refurbishment are taken prior to or at the end of the qualified life of the equipment so that the aging limit is not exceeded.

A.1.36 Boraflex Monitoring

The Boraflex monitoring aging management program is no longer required due to the implementation of the NETCO-SNAP-IN® rack inserts (EC 389362). The rack inserts replace the neutron absorbing function of the Boraflex; therefore, the Boraflex is no longer credited for neutron absorption. The rack inserts are credited for neutron absorption and do not contain Boraflex. Boraflex monitoring has been suspended.

A.1.37 Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrument Circuits

The cables of the Nuclear Instrumentation systems which include Source Range Monitors (SRM's), Intermediate Range Monitors (IRM's), Local Power Range Monitors (LPRM's), and the Radiation Monitoring systems which include Drywell High Range Radiation Monitors, Main Steam Line Radiation Monitors, and the Steam Jet Air Ejector Radiation Monitors are sensitive instrumentation circuits with low-level signals and are located in areas where the cables could be exposed to adverse localized environments caused by heat, radiation, or moisture. These adverse localized environments can result in reduced insulation resistance causing increases in leakage currents. Calibration testing, cable testing or surveillance testing is performed to ensure that the cable insulation resistance is adequate for the instrumentation circuits to perform their intended functions. This provides sufficient indication of the need for corrective actions based on acceptance criteria related to instrumentation loop performance and cable testing. This aging management program is a new program. The calibration testing, cable testing and surveillance testing that will be used for this program are performed currently, and are effective in identifying the existence of age related degradation. The program was implemented prior to the period of extended operation and includes a review of the calibration and surveillance results for cable aging degradation.

A.2 PLANT-SPECIFIC AGING MANAGEMENT PROGRAMS

A.2.1 Corrective Action Program

The 10 CFR Part 50, Appendix B program provides corrective actions, confirmation processes, and administrative controls for aging management programs for license renewal. Prior to the period of extended operation the scope of the program was expanded to include non-safety-related structures and components that are subject to an aging management review for license renewal. The corrective action program applies to all plant systems, structures and components (both safety-related and non-safety-related) within the scope of license renewal. Administrative controls are in place for existing aging management programs and activities. Administrative controls were applied to new and enhanced programs and activities as they were implemented. As a minimum, these programs and activities are performed in accordance with written procedures that are reviewed and approved in accordance with the Quality Assurance Program.

A.2.2 Periodic Inspection of Non-EQ, Non-Segregated Electrical Bus Ducts

This program inspects the non-segregated bus ducts that connect the reserve auxiliary transformers to the 4160V Engineering Safety Systems (ESS) buses for signs of aging degradation that indicate possible loss of intended function. This program was enhanced prior to the period of extended operation to inspect the bus bar insulation material at the accessible bolted connections of the non-segregated bus ducts that connect the reserve auxiliary transformer to the 4160V ESS buses and inspect 10% of the splice insulation material at the bolted connections for the non-segregated bus ducts that connect the EDGs to the ESS buses for signs of aging degradation that indicate possible loose connections. For non-segregated bus ducts that connect the EDGs to the ESS buses, the enhancement included inspections for the presence of dirt or moisture in the bus ducts. The visual inspection now includes all visible insulation in both directions beyond the location of the bolted connection splice insulation inspected.

These bus ducts are in scope of license renewal but are not subject to 10 CFR 50.49 environmental qualification requirements. This inspection program considers the technical information and guidance provided in IEEE Standard P1205, “IEEE Guide for Assessing, Monitoring and Mitigating Aging Effects on Class 1E Equipment Used in Nuclear Power Generating Stations,” SAND 96-0344, “Aging Management Guideline for Commercial Nuclear Power Plants - Electrical Cable and Terminations,” and EPRI TR-109619, "Guideline for the Management of Adverse Localized Equipment Environments."

A.2.3 Periodic Inspection of Ventilation System Elastomers

The periodic inspection of ventilation system elastomers aging management program provides for routine inspections of certain elastomers in the standby gas treatment, emergency diesel generator building ventilation, station blackout diesel generator building ventilation, and main control room ventilation systems. Prior to the period of extended operation an existing program for inspection of ventilation system elastomers was enhanced. The program includes inspections for cracking, loss of material, or other evidence of aging of all flexible boots, access door seals and gaskets, and filter seals and gaskets in the components of these systems that are within the scope of license renewal. The scope of inspections include RTV silicone used as a duct sealant, in systems within the scope of license renewal.

A.2.4 Periodic Testing of Drywell and Torus Spray Nozzles

Carbon steel piping upstream of the drywell and torus spray nozzles is subject to possible general corrosion. The periodic flow tests of drywell and torus spray nozzles address a concern that rust from the possible general corrosion may plug the spray nozzles. These periodic tests verify that the drywell and torus spray nozzles are free from plugging that could result from corrosion product buildup from upstream sources.

A.2.5 Lubricating Oil Monitoring Activities

The lubricating oil monitoring activities aging management program manages loss of material, cracking, and elastomer hardening/loss of strength in lubricating oil heat exchangers and other specific components in the scope of license renewal by monitoring physical and chemical properties in lubricating oil and heat exchanger performance testing. Sampling, testing, and trending verify lubricating oil properties and proper heat exchanger operation. Oil analysis permits identification of specific wear mechanisms, contamination, and oil degradation within operating machinery.

These activities apply to the emergency diesel generator system, station blackout diesel generator system, HPCI system, electro-hydraulic control (EHC) system, reactor core isolation cooling system, and generator hydrogen seal oil system. The complete aging management program for the emergency diesel generator oil coolers, station blackout diesel generator oil coolers, and HPCI oil coolers also includes secondary-side (heat sink) chemistry controls, performance monitoring, and inspections. Those portions of the lubricating oil heat exchanger management program are described in:

- Section A.1.14, “Closed-Cycle Cooling Water System,” for the diesel generator and station blackout diesel generator oil coolers;
- Section A.2.6, “Heat Exchanger Test and Inspection Activities,” for the HPCI oil coolers.

A.2.6 Heat Exchanger Test and Inspection Activities

The heat exchanger test and inspection activities aging management program provides condition monitoring, inspection, and performance testing to manage loss of material, cracking, and buildup of deposits in heat exchangers in the scope of license renewal, that are not tested and inspected under “Open-Cycle Cooling Water” or “Closed-Cycle Cooling Water” aging management programs.

These are new activities that were implemented prior to the period of extended operation.

These activities include tests, inspections, and monitoring and trending of test results to confirm that aging effects are managed. To ensure that system and component functions are maintained, these components are also included in the scope of other activities, which provide inservice inspection and performance monitoring, and primary and secondary-side (water and oil) chemistry controls.

- Management of water chemistry is described in Section A.1.2, “Water Chemistry.”
- Management of the primary, oil side of the HPCI lubricating oil coolers is described in Section A.2.5, “Lubricating Oil Monitoring Activities.”

A.2.7 Generator Stator Water Chemistry Activities

The generator stator water chemistry activities aging management program manages loss of material and cracking aging effects by monitoring and controlling water chemistry. Generator stator water chemistry control maintains high purity water in accordance with General Electric guidelines for stator cooling water systems. Generator stator water is continuously monitored for conductivity and an alarm annunciates if conductivity increases to a predetermined limit.

A.2.8 Periodic Inspection of Plant Heating System

The periodic inspection of plant heating system aging management program provides for routine inspections of selected components in the plant heating system. Prior to the period of extended operation, a new program for periodic inspection of selected components in the plant heating system was implemented. The selected components will be inspected to ensure they are free of cracking, loss of material and leakage. Visual examinations will be conducted by personnel certified to perform VT-3 examinations and will be performed in accordance with approved work instructions. The work instructions will include guidance for identifying the aging management effect of concern.

A.2.9 Periodic Inspection of Components Subject to Moist Air Environments

The periodic inspection of components subject to moist air environments aging management program provides for periodic inspections of selected components exposed to moist air environments and subject to wetting conditions based on system operation. Prior to the period of extended operation, a new program for periodic inspection of selected components was implemented. The inspections consist of UT examinations of components with interior surfaces that are inaccessible and visual inspection (VT-3) of components with accessible interior surfaces for the presence of loss of material due to general corrosion, pitting and crevice corrosion. The visual and UT examinations will be conducted by personnel certified to perform VT-3 and UT examinations accordingly and will be performed in accordance with approved work instructions. The work instructions will include guidance for identifying the aging management effect of concern. In addition, visual inspection of flexible hoses determine any age-related degradation prior to loss of function.

A.2.10 Periodic Inspection of Steam Dryers

Dresden and Quad Cities perform periodic inspections of the steam dryer plate material and welds for evidence of cracking. These inspections are performed in accordance with an NRC approved BWRVIP 139-A.

A.3 TIME-LIMITED AGING ANALYSIS SUMMARIES

In the descriptions of this section, Class I and Class II are the Quad Cities safety classifications described in UFSAR Section 3.2.

A.3.1 Neutron Embrittlement of the Reactor Vessel and Internals

The ferritic materials of the reactor vessel are subject to embrittlement due to high energy neutron exposure. Reactor vessel neutron embrittlement is a TLAA.

A.3.1.1 Reactor Vessel Materials Upper-Shelf Energy Reduction Due to Neutron Embrittlement

The reactor vessel end-of-life neutron fluence has been recalculated for a 60-year (54 EFPY) extended licensed operating period.

The 54 EFPY USE was evaluated by an equivalent margin analysis (EMA) using the 54 EFPY calculated fluence and the Quad Cities surveillance capsule results in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

A.3.1.2 Adjusted Reference Temperature for Reactor Vessel Materials Due to Neutron Embrittlement

The reactor vessel materials peak fluence, $\Delta RT_{NDT,}$ and ART values for the 60-year (54 EFPY) license operating period were calculated in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

A.3.1.3 Reflood Thermal Shock Analysis of the Reactor Vessel

The effects of a reflood thermal shock described in UFSAR Section 3.9.5.3.3 were examined. An alternative analysis confirms that the effects remain acceptable for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

A.3.1.4 Reflood Thermal Shock Analysis of the Reactor Vessel Core Shroud and Repair Hardware

Radiation embrittlement may affect the ability of reactor vessel internals, particularly the core shroud and repair hardware, to withstand a low-pressure coolant injection (LPCI) thermal shock transient. Embrittlement effects are evaluated for the maximum-fluence beltline region of the core shroud, where the maximum event strain is about 0.57 percent [UFSAR Section 3.9.5.3.2], and design of the core shroud repair tie rod stabilizer assemblies included an investigation of possible embrittlement effects.

The effects of the increase in neutron fluence with a 54 EFPY life at uprated power were evaluated, and the allowable strain for this faulted event remains a considerable margin above the expected strain.

The core shroud repair tie rod stabilizer assemblies were designed for a 40-year life, which will not be exceeded at the end of the extended licensed operating period.

The existing analyses of the effects of embrittlement in the internals have been evaluated and remain valid for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

A.3.1.5 Reactor Vessel Thermal Limit Analyses: Operating Pressure – Temperature Limits

Revised pressure-temperature (P-T) limits for a 60-year licensed operating life (54 EFPY) have been submitted to and approved by the NRC.

A.3.1.6 Reactor Vessel Circumferential Weld Examination Relief

Relief has been requested from the requirements for inspection of RPV circumferential welds for the remainder of the current 40-year licensed operating period. The justification for relief is consistent with Boiling Water Reactor Vessel and Internals Program BWRVIP-05, “BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations,” guidelines. Application for an extension of this relief is included in the fifth interval ISI update.

The procedures and training that will be used to limit the frequency of cold over-pressure events to the number specified in the SER for the RPV circumferential weld relief request extension, during the license renewal term, are the same as those approved for use in the current period (Ref. 3 and 4).

The analyses associated with reactor vessel circumferential weld examination relief has been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

A.3.1.7 Reactor Vessel Axial Weld Failure Probability

BWRVIP-05, “BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations,” estimated the 40-year end-of-life failure probability of a limiting reactor vessel axial weld, showed that it was orders of magnitude greater than the 40-year end-of-life circumferential weld failure probability, and used this analysis to justify relief from inspection of the circumferential welds, as described in Section A.3.1.6.

The re-evaluation of the axial weld failure probability for 60 years depends on vessel ΔRT_{NDT} calculations. The NRC staff review and the NRC staff and BWRVIP calculations of the test-case failure probabilities assume that 90 percent of axial welds will be inspected. At Quad Cities, less than 90 percent of axial welds can be inspected. As such, an analysis was performed for 54 EFPY to assess the effect on the probability of fracture due to the actual inspection performed on the vessel axial welds and to determine if the coverage was sufficient in the inspection of regions contributing to the majority of the risk.

The evaluation shows that the calculated unit-specific axial weld conditional failure probabilities at 54 EFPY for Quad Cities are less than the failure probabilities calculated by the NRC staff in the NRC BWRVIP-05 SER at 64 EFPY and the limiting Clinton values found in Table 3 of the SER supplement. The projected probability of failure of an axial weld at Quad Cities will therefore provide adequate margin above the probability of failure of a circumferential weld, in support of relief from inspection of circumferential welds, for the extended licensed operating period, in accordance with the requirements of 10 CFR 54.21 (c)(1)(ii).

A.3.2 Metal Fatigue

The thermal and mechanical fatigue analyses of mechanical components have been identified as TLAAs for Quad Cities. Specific components have been designed considering transient cycle assumptions, as listed in vendor specifications and the Quad Cities UFSAR.

A.3.2.1 Reactor Vessel Fatigue

Unit 1 and Unit 2 reactor vessel fatigue analyses depend on cycle count assumptions that assume a 40-year operating period. The effects of fatigue in the reactor vessel will be managed for the period of extended operation by the fatigue management program for cycle counting and fatigue usage factor tracking, as described in Section A.1.34.

This aging management program will ensure that fatigue effects in vessel pressure boundary components will be adequately managed and will be maintained within code design limits for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

A.3.2.2 Fatigue Analysis of Reactor Vessel Internals

A.3.2.2.1 Low-Cycle Thermal Fatigue Analysis of the Core Shroud and Repair Hardware

Only one Quad Cities analysis of low-cycle fatigue in RPV internals exists: the evaluation of a standard design for repair of the core shroud. This analysis is a TLAA. The calculated fatigue effects are not significant.

The fatigue analysis of the core shroud repair has been evaluated and remains valid for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

A.3.2.3 Reactor Coolant Pressure Boundary Piping and Component Fatigue Analysis

A.3.2.3.1 Reactor Coolant Pressure Boundary Piping and Components Designed to USAS B31.1, ASME Section III Class 2 and 3, or ASME Section VIII Class B and C

All primary system and other reactor coolant pressure boundary (RCPB) piping systems were designed to USAS B31.1, 1967 Edition, as were the safety relief valve (SRV) discharge lines inside the drywell. The USAS B31.1 piping design does not invoke a fatigue analysis, but USAS B31.1 does apply a stress range reduction factor based on an assumed finite number of equivalent full-range thermal cycles for the design life. The B31.1 designs are therefore TLAAs because they are part of the current licensing basis, are used to support a safety determination, and depend on a specific number of cycles which might change with a change in licensed operating life.

The assumed number of design lifetime equivalent full-range thermal cycles determines the allowable stress range (the stress range reduction factor) for design of all Class I and Class II USAS B31.1 or ASME Class 2 or 3 piping. With the exception of containment vent and process bellows, no components in the scope of license renewal designed to ASME Section III or Section VIII require design for cyclic thermal loading. The number of thermal cycles assumed for design of Class I and II piping has been evaluated and the existing stress range reduction factor remains valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

A.3.2.4 Effects of Reactor Coolant Environment on Fatigue Life of Components and Piping (Generic Safety Issue 190)

Generic Safety Issue (GSI) 190 was identified by the NRC because of concerns about potential effects of reactor water environments on component fatigue life during the period of extended operation.

Prior to the period of extended operation, Exelon performed plant-specific calculations for the applicable locations identified in NUREG/CR 6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," for older-vintage BWR plants, to assess the potential effects of reactor coolant on component fatigue life in accordance with 10 CFR 54.21(c)(1)(ii). The calculations of current and projected cumulative usage factors (CUFs) under this program will include appropriate environmental fatigue effect (F_{EN}) factors from NUREG/CR 6583 and NUREG/CR 5704. Appropriate corrective action will be taken if the resulting projected end-of-life CUF values exceed 1.0.

Exelon reserves the right to modify this position in the future based on the results of industry activities currently underway, or based on other results of improvements in methodology, subject to NRC approval prior to changes in this position.

A.3.3 <u>Environmental Qualification of Electrical Equipment</u>

Electrical equipment included in the Quad Cities Environmental Qualification Program which has a specified qualified life of at least 40 years involves time-limited aging analyses for license renewal. The aging effects of this equipment will be managed in the Environmental Qualification Program discussed in Section A.1.35, “Environmental Qualification (EQ) of Electrical Components,” in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

A.3.4 <u>Containment Fatigue</u>

The Quad Cities Mark I containments were originally designed to stress limit criteria without fatigue analyses. However, the discovery of significant hydrodynamic loads (“new loads”) caused by safety relief valve (SRV) and small, intermediate, and design basis pipe break discharges into the suppression pool required the reanalysis of the suppression chamber, vents, and attached piping and internal structures, including some fatigue analyses at limiting locations. These fatigue analyses of the suppression chamber, and its internals, and vents in each unit include assumed pressure, temperature, seismic, and SRV cycles, and combinations thereof. The scope of the analyses included the suppression chamber, the drywell-to-suppression chamber vents, SRV discharge piping, other piping attached to the suppression chamber and its penetrations, and the drywell-to-suppression chamber vent bellows.

A.3.4.1 Fatigue Analysis of the Suppression Chamber, Vents, and Downcomers

For low cumulative usage factor (CUF) locations (40-year CUF < 0.4) the Quad Cities new loads analyses of each suppression chamber and its associated vents and downcomers have been evaluated and remain valid for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

For higher cumulative usage factor locations in the analyses of the suppression chamber and suppression chamber vents and downcomers (40-year CUF ≥ 0.4) the effects of fatigue will be managed for the period of extended operation by the fatigue management cycle counting and fatigue usage factor tracking program, as described in Section A.1.34.

The fatigue management activities will ensure that fatigue effects in containment pressure boundary components are adequately managed and are maintained within code design limits for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

A.3.4.2 Fatigue Analysis of SRV Discharge Piping Inside the Suppression Chamber, External Suppression Chamber Attached Piping, and Associated Penetrations

SRV discharge lines and external suppression chamber attached piping and associated penetrations were analyzed separately from the suppression chamber, vents and downcomers. The disposition of these analyses is the same as described for the suppression chamber, vents and downcomers in Section A.3.4.1 above.

A.3.4.3 Drywell-to-Suppression Chamber Vent Line Bellows Fatigue Analyses

A fatigue analysis of the drywell-to-suppression chamber vent line bellows was performed assuming 150 thermal and internal pressure load cycles for the 40-year life of the plant. The drywell-to-suppression chamber vent line bellows have a rated capacity of 1,000 cycles at maximum displacement.

The Quad Cities new loads fatigue analysis of the drywell-to-suppression chamber vent line bellows have been evaluated and remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

A.3.4.4 Primary Containment Process Penetrations Bellows Fatigue Analysis

The only containment process piping expansion joints subject to significant thermal expansion and contraction are those between the drywell shell penetrations and process piping. These are designed for a stated number of operating and thermal cycles.

The thermal cycle designs of Quad Cities containment process penetration bellows have been evaluated and remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

A.3.5 Other Plant-Specific TLAAs

A.3.5.1 Reactor Building Crane Load Cycles

The reactor building overhead cranes in Quad Cities were designed to meet or exceed the design criteria of the Crane Manufacturers Association of America (CMAA) Specification 70, "Specifications for Electric Overhead Traveling Cranes," Class A1. These cranes are capable of a minimum of 100,000 cycles at the full rated load of 125 tons. Correspondence with the NRC stated that over their 40-year life these cranes would most probably see fewer than 5,000 cycles at a maximum of 100 tons, and a larger number of cycles at significantly less than 100 tons.

The load cycle designs of Quad Cities reactor building cranes have been evaluated and remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

A.3.5.2 Metal Corrosion Allowances

A.3.5.2.1 Corrosion Allowance for Power Operated Relief Valves

GE specification 25A5508, "Relief Valve, Power Operated," for the Quad Cities Unit 2 replacement PORVs prescribes a corrosion allowance of 0.002 inches for stainless steel and 0.120 inches for carbon steel for a design life of 40 years. The specification is cited in Quad Cities UFSAR Section 5.2.2.

The corrosion allowance for the Quad Cities Unit 2 replacement PORVs has been evaluated and remains valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

A.3.5.2.2 Degradation Rates of Inaccessible Exterior Drywell Plate Surfaces

In its response to Generic Letter 87-05, "Request for Additional Information Assessment of Licensee Measures to Mitigate and/ or Identify Degradation of Mark I Drywells," Commonwealth Edison evaluated the potential effects of corrosion on exterior drywell steel surfaces in the "sand pockets" of Dresden Unit 3 drywell and found that 27 years of service remained before corrosion at the assumed rate would have a significant adverse effect on design basis stresses. The evaluation concluded that the findings were applicable to Dresden Unit 2 and Quad Cities Units 1 and 2 as well.

A program was instituted for the Dresden Unit 3 inaccessible annulus areas to monitor potential corrosion. Dresden Unit 3 is considered the limiting case for potential drywell corrosion among the four Dresden and Quad Cities units.

The program will inspect a sample of locations in the cylindrical and upper spherical areas of the drywell, using ultrasonic measurements of the drywell shell thickness made from accessible areas of the drywell interior. A baseline inspection was performed prior to the period of extended operation. A follow-up inspection consisting partly of the same locations and partly of variable locations will be conducted by the third refueling outage after the baseline inspection. The follow-up inspection will be used to determine whether any corrosion is occurring and that any observed corrosion rate will not threaten drywell integrity during the extended 60-year plant life.

A.3.5.2.3 Galvanic Corrosion in the Containment Shell and Attached Piping Components due to Stainless Steel ECCS Suction Strainers

The Quad Cities ECCS suction strainers have been replaced with larger strainers. The replacement strainers are stainless steel. The modification included drilling new bolt holes and enlarging the existing bolt holes in each of the existing carbon steel strainer support flanges to provide sufficient bolting for the larger replacement strainers. The holes in the carbon steel flanges are not coated to protect them from corrosion. The calculation of corrosion effects assumes a corrosion allowance of 4 mils/year and assumes a design life of 33 years, which is just short of the 60-year extended operating period.

The corrosion rate assumptions used in the calculation will be monitored by an ultrasonic inspection at Dresden Station. Dresden Station will continue UT thickness measurements for Unit 2 and Unit 3 on the same flange bolt hole every other refuel outage and will evaluate the UT data establishing the corrosion rate to validate acceptable wall thickness to the end of the 60-year licensed operating period, in accordance with 10 CFR 54.21(c)(1)(ii). In the event that the measured galvanic corrosion rate will not ensure acceptable thickness to the end of the 60-year licensed operating period, appropriate corrective action will be identified and implemented to maintain the structural integrity of the strainer flanges.

A.3.5.3 Crack Growth Calculation of a Postulated Flaw in the Heat Affected Zone of an Arc Strike in the Suppression Chamber Shell

A calculation provides technical justification for continued operation of the Quad Cities Unit 2 torus which was damaged by an arc strike. The flaw has been ground smooth and NDE tested. It was initially assumed the damaged area would be repaired after two fuel cycles of operation. This time limit has been extended with appropriate NDE being performed to assure no cracks or other linear flaws exist in the affected area.

The crack growth calculation has been evaluated and remains valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

A.3.5.4 Radiation Degradation of Drywell Shell Expansion Gap Polyurethane Foam

The steel drywell shell is largely enclosed within the structural and shielding concrete of the reactor containment building. To accommodate thermal expansion, compressible foam was used to form an expansion gap between the concrete and the drywell shell. A confirming analysis contained in the UFSAR evaluates the increase in external compressive loads on the drywell exterior, due to additional compression of this foam, for accident-condition thermal expansion of the drywell. The load depends on the stress-strain curve of the foam, and the validity of this confirming analysis of the Quad Cities drywells therefore depends on the stiffness of the polyurethane foam. The analysis would require validation if the foam became stiffer (higher compressive stress for the same strain) as a result of increased radiation exposure from extended plant operation.

The expected radiation exposure of the foam has been evaluated and remains below the significant damage threshold at the end of the period of extended operation. The evaluation of thermal expansion compressive loads therefore also remains valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

A.3.6 <u>References for Section A.3</u>

1. Dresden Nuclear Power Station Units 2 and 3, Quad Cities Nuclear Power Station Units 1 and 2, License Renewal Project, TLAA Technical Report. Revision 0, June 2002. Prepared by Parsons Energy and Chemicals, Inc. for the General Electric Company.

2. Dresden Nuclear Power Station Units 2 and 3, Quad Cities Nuclear Power Station Units 1 and 2, License Renewal Project, Potential TLAA Review Results Package. Revision 0, June 2002. Prepared by Parsons Energy and Chemicals, Inc. for the General Electric Company.

3. Quad Cities Letter RS-03-099 from Patrick R. Simpson (Exelon) to USNRC, Relief Request for Alternative Reactor Pressure Vessel Circumferential Weld Examinations for Fourth Interval Inservice Inspection Program, Letter dated May 16, 2003.

4. Quad Cities Letter RS-03-131 from Patrick R. Simpson (Exelon) to USNRC, Additional Information Supporting the Relief Request for Alternative Reactor Pressure Vessel Circumferential Weld Examinations for Fourth Interval Inservice Inspection Program, Letter dated July 7, 2003

A.4 NEWLY IDENTIFIED SSCs (10 CFR 54.37(b))

After the renewed license is issued, the UFSAR update required by 10 CFR 50.71(e) must include any systems, structures, and components newly identified that would have been subject to an aging management review or evaluation of time-limited aging analyses in accordance with §54.21. This UFSAR update must describe how the effects of aging will be managed such that the intended function(s) in §54.4(b) will be effectively maintained during the period of extended operation.

No.	Date Identified	SSC Description	Aging Management Review (AMR) Conclusion	Aging Management Program
1.	09/22/2011	The Environmental Qualification (EQ) Program was revised to reflect increases in the assumed zone temperatures and pressures in the first floor of the Reactor Building. These changes resulted in the addition of several SSCs into the EQ Program. These SSCs were installed at the time of the license renewal review, so they are considered “newly identified” under the terms of 10 CFR 54.37(b). The changes to the EQ Program zone assumptions associated with the Reactor Building components were noted in a revision to UFSAR Figure 3.11-1, Sheets 1a, 1b, 1c, 1d, 3a, 5a, 7a. Specific SSCs impacted by this change are noted in EC 370997.	SSCs that are included in the scope of the EQ Program are subject to Time Limited Aging Analysis (TLAA). Since these SSCs have been added to the scope of the EQ Program, they are within the scope of its TLAA.	The existing EQ Program was credited without revision as Aging Management Program (AMP) B.1.35 in the Renewed License SER (NUREG 1796). The addition of these newly identified SSCs to the scope of the EQ Program also included them in the scope of Aging Management Program (AMP) B.1.35. These SSCs are now subject to the same aging management activities as those in the License Renewal Application.