APPENDIX O

AGING MANAGEMENT PROGRAMS

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O.0 INTRODUCTION

This appendix contains a summary description of the Aging Management Program activities and the Time-Limited Aging Analyses (TLAAs) required for license renewal. These summary descriptions have been incorporated into the Updated Final Safety Analysis Report for the Browns Ferry Nuclear Plant in preparation for the period of extended operation. This appendix also includes newly identified systems, structures, and components as required by 10 CFR 54.37(b).

The integrated plant assessment for license renewal identified new programs, enhancements to existing programs, and existing programs necessary to continue operation of BFN Units 1, 2, and 3 during the additional twenty years beyond the initial license term. This chapter describes those programs. The TVA Nuclear Quality Assurance Program implements the requirements of 10 CFR 50, Appendix B. The TVA Nuclear Quality Assurance Program includes elements of corrective action, confirmation process, and administrative controls. These elements are applicable to all aging management programs credited for license renewal including programs for safety-related and nonsafety-related structures, systems, and components. The Corrective Action Program ensures corrective actions, including root cause determinations and prevention of recurrence are timely. The Corrective Action Program also includes the confirmation process that ensures preventive actions are adequate and that appropriate corrective actions have been complete and effective. Administrative controls provide for a formal review and approval process of program implementation documents.

O.1 AGING MANAGEMENT PROGRAMS

O.1.1 Accessible Non-Environmental Qualification Cables and Connections Inspection Program

The Accessible Non-Environmental Qualification (EQ) Cables and Connections Inspection Program manages the aging effects of insulated cables and connections within the scope of license renewal exposed to adverse localized environments. The Accessible Non-EQ Cables and Connections Inspection Program is a condition monitoring program. A sample of non-EQ insulated cables and connections are periodically inspected. The sample includes power, instrumentation, control, and communication applications located in accessible adverse localized environments.

The scope of the program includes a review of operating experience. As a minimum the sample includes a representative sample of accessible cable and connections that are not coated with Flamemastic. The sample includes cables in the Drywell and Turbine Building.

If an unacceptable condition or situation is identified for a cable or connection in the inspection sample, an evaluation will determine if the same condition or situation is applicable to other accessible or inaccessible cables or connections.

O.1.2 Electrical Cables Not Subject To 10 CFR 50.49 Environmental Qualification Requirements Used In Instrumentation Circuits Program

The Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification (EQ) Requirements Used in Instrumentation Circuits Program manages the aging effects of sensitive, low level signal circuits exposed to adverse localized environments. The scope of this program is limited to the cables of the Intermediate Range Monitors and Local Power Range Monitors.

The Intermediate Range Monitor cable system portion of this program is a condition monitoring program which performs direct testing of the cable system for detecting deterioration of the insulation system. When intermediate range monitor cable system testing results are not within the limits of the acceptance criteria, evaluations that consider the possibility of cable degradation are performed and appropriate corrective actions are taken if cable degradation is determined to exist.

The Local Power Range Monitor cable system portion of this program is a condition monitoring program using the calibration test of the local power range monitors mandated by Technical Specifications. When a local power range monitor is found to be out of calibration, evaluations that consider the possibility of cable degradation are performed and appropriate corrective actions are taken if cable degradation is determined to exist.

O.1.3 Inaccessible Medium Voltage Cables, Not Subject To 10 CFR 50.49 Environmental Qualification Requirements Program

The Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program manages the aging effects of medium voltage cables that are not subject to the environmental qualification requirements of 10 CFR 50.49 and are exposed to adverse localized environments caused by moisture while energized. The program for Inaccessible Medium Voltage Cables, Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program, is a condition monitoring program. Medium voltage cables that are installed in underground conduit duct banks and that perform an intended function within the scope of license renewal are periodically tested to provide an indication of the condition of the conductor insulation. The type of test performed is a proven test for detecting deterioration of cable insulation due to wetting. Additionally, handholes and manholes associated with in-scope cables are visually inspected for signs of moisture on a periodic basis.

O.1.4 ASME SECTION XI INSERVICE INSPECTION SUBSECTIONS IWB, IWC, AND IWD PROGRAM

10 CFR 50.55(a) imposes the inservice inspection requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section XI for Class 1, 2, and 3 pressure retaining components and their integral attachments. The ASME Section XI Subsections IWB, IWC, and IWD Inservice Inspection Program includes periodic visual, surface, and/or volumetric examination of Class 1, 2, and 3 pressure retaining components and their integral attachments.

O.1.5 Chemistry Control Program

The Chemistry Control Program consists of monitoring and control of water chemistry to keep peak levels of various contaminants below system specific limits based on the applicable revision of EPRI BWR Water Chemistry Guidelines as referenced in associated plant procedures.

O.1.6 Reactor Head Closure Studs Program

The BFN Reactor Head Closure Studs Program includes:

(a) Inservice inspection in conformance with the requirements of the American Society of Mechanical Engineers B&PV Code Section XI, Subsection IWB, Table IWB 2500-1.

(b) Preventive measures to mitigate cracking. The preventive measures of Regulatory Guide 1.65, “Materials and Inspections for Reactor Vessel Closure Studs,” are implemented. Approved lubricants minimize the potential of cracking of the non-metal-plated reactor head closure studs.

O.1.7 Boiling Water Reactor Vessel Inside Diameter Attachment Welds Program

The Boiling Water Reactor (BWR) Vessel Inside Diameter (ID) Attachment Welds 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.” Inspections and flaw evaluations are performed by the ASME B&PV Code Section XI, Subsections IWB, IWC, and IWD aging management program.

(b) Monitoring and control of reactor coolant water chemistry by the Chemistry Control Program.

O.1.8 Boiling Water Reactor Feedwater Nozzle Program

The BWR Feedwater Nozzle Program includes:

(a) Inservice inspection in accordance with the requirements of the ASME B&PV Code Section XI, Subsection IWB, Table IWB 2500-1 and the recommendations of General Electric NE-523-A71-0594-A, Rev. 1 “Alternate BWR Feedwater Nozzle Inspection Requirements."

(b) System modifications to mitigate cracking. The program addressed BWR feedwater nozzle cracking by implementation of NUREG-0619, “BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking,” recommendations. The BFN feedwater nozzles have been modified to mitigate cracking by removing the stainless steel cladding and machining the safe end and nozzle bore and inner bend radius to accept improved double piston ring interference fit spargers with a forged tee design and orificed elbow discharges. The reactor water cleanup system return lines were routed to both feedwater headers (Unit 3 only). Changes to plant operating procedures, such as improved feedwater control, to decrease the magnitude and frequency of temperature fluctuations have been implemented.

O.1.9 Boiling Water Reactor Control Rod Drive Return Line Nozzle Program

The BWR Control Rod Drive Return Line Nozzle Program includes:

(a) Inservice inspection in accordance with the ASME B&PV Code Section XI, Subsection IWB, IBC, and IWD aging management program.

(b) System modifications to mitigate cracking. BFN has modified the Control Rod Drive Return lines to meet the recommendations of NUREG-0619, “BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking.” The Control Rod Drive Return line flow returns to the Reactor Water Cleanup System piping. The Control Rod Drive Return line nozzle piping has been removed and the reactor vessel nozzles have been capped.

O.1.10 Boiling Water Reactor Stress Corrosion Cracking Program

The BWR Stress Corrosion Cracking Program manages intergranular stress corrosion cracking in reactor coolant pressure boundary components made of stainless steel. The BWR Stress Corrosion Cracking Program is consistent with NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," BWRVIP-75, “Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules,” and Nuclear Regulatory Commission Generic Letter 88-01, “NRC Position on Intergranular Stress Corrosion

Cracking in BWR Austenitic Stainless Steel Piping," and its Supplement 1. The program includes:

(a) Replacements and preventive measures to mitigate intergranular stress corrosion cracking. Replacement methodologies include piping replacement with IGSCC resistant stainless steel. Preventive measures include heat sink welding, induction heating, mechanical stress improvement and water chemistry control in accordance with industry recognized guidelines.

(b) Inspections to monitor intergranular stress corrosion cracking and its effects. The ASME B&PV Code Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program has incorporated the guidelines delineated in NUREG-0313, NRC GL 88-01, and BWRVIP-75.

(c) Some welds within the scope of the Boiling Water Reactor Stress Corrosion Cracking Program cannot be fully examined due to access limitations caused by design, geometry, or materials of construction of the components. Limitations are typically encountered in austenitic stainless steel and dissimilar metal welds to cast components, valves, and fittings where access to the weld for examination from both sides is restricted. Ultrasonic examinations (UT) will be performed to the maximum extent practical due to the configuration and design using the latest ultrasonic techniques, procedures, equipment, and personnel qualified to the requirements of the Performance Demonstration Initiative (PDI) Program in accordance with 10 CFR 50.55a(g)(4) and 10 CFR 50.55a(g)(6)(ii)(C). Typically, the IGSCC susceptible side of the welds is adequately examined and the limitation is encountered on the cast (non-IGSCC susceptible) side of the weld. These welds are part of a larger population of welds examined in which examination coverage is not reduced and for which the required coverage is attained. When considered in aggregate with the entire sample population, an adequate level of inspection occurs to provide reasonable assurance that a pattern of IGSCC degradation that, if present, could affect the overall integrity of the components would be detected.

O.1.11 Boiling Water Reactor Penetrations Program

The BWR Penetrations 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. Inspection and flaw evaluation is conducted in accordance with the ASME B&PV Code Section XI Subsections IWB, IWC, and IWD Inservice Inspection Program and the augmented exam recommendations of the BWRVIP-27 and BWRVIP-49 guidelines. Required repairs or replacements implement the recommendations of BWRVIP-53, “Standby Liquid Control Line Repair Design Criteria,” and BWRVIP-57, “Instrument Penetration Repair Design Criteria,” into procedures which are performed in accordance with ASME Section XI repair and replacement requirements.

(b) Monitoring and control of reactor coolant water chemistry in accordance with guidelines of EPRI, “BWR Water Chemistry Guidelines.”

O.1.12 Boiling Water Reactor Vessel Internals Program

The BWR Vessel Internals Program includes:

(a) Inspection and flaw evaluation in conformance with the guideline of applicable Boiling Water Reactor Vessel and Internals Project (BWRVIP) documents. Additionally, certain components that are addressed as part of the BWR Vessel Internals Program are inspected in accordance with ASME Section XI. The processes of BWRVIP-94, “BWR Vessel and Internals Project Program Implementation Guide,” are used to implement the BWR Vessel Internals Program at BFN.

(b) Monitoring and control of reactor coolant water chemistry in accordance with industry recognized guidelines per applicable revision of EPRI, “BWR Water Chemistry Guidelines” as referenced in associated plant procedures to ensure the long term integrity and safe operation of boiling water reactor vessel internal components.

O.1.13 (Deleted)

O.1.14 Flow-Accelerated Corrosion Program

The Flow-Accelerated Corrosion Program consists of appropriate analysis and baseline inspections followed by the determination of the extent of thinning with replacement or repair of components if necessary. This program is in response to NRC Generic Letter 89-08, “Erosion/Corrosion-Induced Pipe Wall Thinning.” The program is based on EPRI NSAC-202L, “Recommendations for an Effective Flow-Accelerated Corrosion Program,” guidelines.

O.1.15 Bolting Integrity Program

The Browns Ferry Bolting Integrity Program provides for condition monitoring of pressure retaining bolting within the scope of license renewal. The Bolting Integrity Program provides for:

(a) Preventive Actions - Plant procedures specify selection of bolting material and the use of lubricants and sealants. The program is consistent with the guidelines of EPRI NP-5769, “Degradation and Failure of Bolting in Nuclear Power Plants,” and the additional recommendations of NUREG-1339, “Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants,” to prevent or mitigate degradation and failure of safety-related bolting.

(b) Condition monitoring - The BFN Bolting Integrity Program includes inservice inspections of Class 1, 2, and 3 components in accordance with ASME Section XI, Subsections IWB, IWC, and IWD Program. Inspection for bolting within the scope of license renewal not included in the ASME Section XI, Inservice Inspection (ISI) aging management program is provided by the System Monitoring Program.

The aging effects of reactor vessel internal bolting and reactor vessel closure bolting are not managed by the Bolting Integrity Program.

O.1.16 Open-Cycle Cooling Water System Program

The Open-Cycle Cooling Water Program manages loss of material, biofouling, pitting, flow blockage, and reduction of heat transfer aging effects in raw cooling water piping and components. The activities for managing aging effects include:

(a) Condition monitoring - System and component testing, visual inspections, and NDE testing are conducted.

(b) Preventive actions - Chemical treatments, filtering, inspections, and cleaning are used to prevent loss of material due to MIC and biofouling and flow blockage and reduction of heat transfer due to biological and particulate fouling.

The Open-Cycle Cooling Water System Program relies on the guidelines of NRC Generic Letter 89-13, “Service Water System Problems Affecting Safety-Related Equipment.”

In addition to the requirements of GL 89-13, the Open-Cycle Cooling Water System Program will be enhanced to perform inspections on the internal portion of one of the embedded RHRSW pipes that run between the CCW Pump Pits to the EECW/RHRSW Pump Pits, the RHRSW sluice gate valves located in the CCW Pump Pits, and the seismic restraints in the RHRSW Pump Pits. These inspections will be performed prior to the expiration of the current 40-year license and will be conducted at least one additional time within 10 years of entering the period of extended operation.

O.1.17 Closed-Cycle Cooling Water System Program

The Closed-Cycle Cooling Water System Program includes (a) preventive measures to minimize corrosion and (b) surveillance testing and inspection to monitor the effects of corrosion on the intended function of the component. The program relies on maintenance of system corrosion inhibitor concentrations within specified limits of applicable revision of EPRI Closed-Cycle Cooling Water Chemistry Guideline as referenced in associated plant procedures to minimize corrosion. Testing and inspection in accordance with applicable revision of EPRI Closed-Cycle Cooling Water Chemistry Guideline as referenced in associated plant procedures is performed to evaluate system and component performance.

O.1.18 Inspection of Overhead Heavy Load and Light Load Handling Systems Program

The Inspection of Overhead Heavy Load and Light Load Handling Systems Program is a condition monitoring program primarily concerned with structural components that make up the bridge and trolley. Visual inspections verify the structural integrity of crane components. Inspection requirements are consistent with the guidance provided by NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants” for load handling systems that handle heavy loads which can directly or indirectly cause a release of radioactive materials and with applicable industry standards for other cranes within the scope of license renewal. The aging management activities in this program utilize the guidance provided in American Society of Mechanical Engineers (ASME) B30 Safety Standards.

O.1.19 Compressed Air Monitoring Program

The Compressed Air Monitoring Program consists of:

(a) Condition monitoring - Inspection and testing of the system is performed.

(b) Preventive actions - Air quality at various locations in the system is monitored to ensure that oil, water, rust, dirt, and other contaminants are kept within the specified limits.

The Compressed Air Monitoring Program is based on NRC GL 88-14, “Instrument Air Supply System Problems Affecting Safety-Related Equipment,” and the Institute of Nuclear Power Operations Significant Operating Experience Report 88-01, “Instrument Air System Failures.” The Compressed Air Monitoring Program also incorporates provisions conforming to the guidance of the EPRI NP-7079, “Instrument Air Systems, A Guide for Power Plant Maintenance Personnel.”

This program incorporates the guidelines of ASME OM-S/G-2000 Part 17, “Performance Testing of Instrument Air Systems in Light-Water Reactor Power

Plants," ANSI/ISA-S7.0.01-1996, "Quality Standard for Instrument Air," and EPRI TR-108147, "Compressor and Instrument Air System Maintenance Guide."

O.1.20 <u>BWR Reactor Water Cleanup System Program</u>

The BWR Reactor Water Cleanup System Program applies to RWCU piping welds outboard of the second isolation valve. No inspections on these welds are required since the piping is made of IGSCC resistant material and requirements of NRC GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," have been implemented. The BWR Reactor Water Cleanup System program monitors and controls reactor water chemistry based on industry recognized guidelines of applicable revision of EPRI BWR Water Chemistry Guidelines as referenced in associated plant procedures. Maintaining the water chemistry to EPRI BWR Water Chemistry Guidelines will reduce the susceptibility of Reactor Water Cleanup System piping to stress corrosion cracking and intergranular stress corrosion cracking.

O.1.21 <u>Fire Protection Program</u>

The Fire Protection Program includes fire barrier inspections and diesel-driven fire pump tests. Fire Protection inspections and tests are documented in the Fire Protection Requirements Manual (FPRM).

The FPRM requires periodic visual inspection of fire barrier penetration seals, fire barrier walls, ceilings, and floors, and periodic visual inspection of fire rated doors to ensure that their operability is maintained. The FPRM requires that the diesel driven fire pumps be periodically tested to ensure that the fuel supply line can perform its intended function. The FPRM also includes periodic inspection and test of the carbon dioxide fire suppression system.

O.1.22 <u>Fire Water System Program</u>

The Fire Water System Program applies to water based fire protection systems that consist of sprinklers, nozzles, fittings, valves, hydrants, hose stations, standpipes, water storage tanks, and aboveground and underground piping and components that are tested in accordance with the applicable National Fire Protection Association codes and standards. The fire water system tests are mandated by the FPRM.

This program includes periodic flow testing and the option to perform non-intrusive examinations to identify loss of material due to corrosion. Sprinkler heads for each unit will be replaced or inspected. If sprinklers are not replaced, they shall be inspected prior to the end of the fifty year service life and at ten year intervals thereafter in accordance with NFPA-25, "Standard for the Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems."

O.1.23 Aboveground Carbon Steel Tanks Program

The Aboveground Carbon Steel Tanks Program includes preventive measures to mitigate corrosion by protecting the external surface of the unit specific condensate storage carbon steel tanks with paint or coatings in accordance with standard industry practice. The Aboveground Carbon Steel Tanks Program also relies on periodic inspections conducted in accordance with the 10 CFR 50.65 Maintenance Rule Program and the Systems Monitoring Program to monitor the unit specific condensate storage tank degradation.

O.1.24 Fuel Oil Chemistry Program

The Fuel Oil Chemistry 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 fuel oil sampling and analysis, including analysis of new fuel before its' introduction into the storage tanks. Sampling and testing of diesel fuel oil is in accordance with American Society for Testing Materials Standards D 1796, "Standard Test Method for Water and Sediment in Fuel Oils by the Centrifuge Method (Laboratory Procedure)," D 2276, "Standard Test Method for Particulate Contamination in Aviation Fuel by Line Sampling," and D 4057, "Standard Practice for Manual Sampling of Petroleum and Petroleum Products," (applicable revision of ASTM Standards as referenced in associated plant procedures). .

O.1.25 Reactor Vessel Surveillance Program

The BFN Reactor Vessel Surveillance Program is mandated by 10 CFR 50, Appendix H. The BFN Reactor Vessel Surveillance Program is an integrated surveillance program (ISP) in accordance with 10 CFR Part 50, Appendix H, Paragraph III.C that is based on requirements established by the BWR Vessel and Internals Project reports.

This program has been enhanced to implement BWRVIP-116 which ensures that the BFN Units 1, 2, and 3 reactor vessels meet the requirements of 10 CFR 50 Appendix H, as discussed in UFSAR Section 4.2.6, Inspection and Testing.

The BFN Units 1 and 3 surveillance capsules (standby capsules) will remain in place and will continue to be irradiated during plant operation, including the period of extended operation. Therefore, the BFN Units 1 and 3 irradiated material samples continue to remain available to the ISP, if needed. If any surveillance capsules are removed without the intent to test them, these capsules will be stored in manner which maintains them in a condition which would support re-insertion into the reactor vessel, if necessary.

O.1.26 One-Time Inspection Program

The One-Time Inspection Program includes measures to verify that unacceptable degradation is not occurring; thereby validating the effectiveness of existing programs or confirming that there is no need to manage aging related degradation for the period of extended operation. The One-Time Inspection Program will be completed prior to entering the period of extended operation. The elements of the One-Time Inspection Program include:

(a) Determination of the sample size based on an assessment of materials of fabrication, environment, plausible aging effects, and operating experience.

(b) Identification of the inspection locations in structures or components based on the aging effect.

(c) Determination of the examination technique, including acceptance criteria effective in detecting the age related effect for which the component is examined. Non-destructive techniques are, in general, used.

(d) Evaluation of the need for follow up examinations to monitor the progression of any aging degradation. When one-time inspections fail to meet the established acceptance criteria, the corrective action program is used to schedule, track, and trend the appropriate corrective actions and follow up inspections.

O.1.27 Selective Leaching of Materials Program

The Selective Leaching of Materials Program consists of one-time visual inspections coupled with either hardness measurements or mechanical examination techniques on selected components susceptible to selective leaching. The materials of construction for these components may include cast iron, brass, bronze, or aluminum-bronze. These components may be exposed to a raw water, treated water, or ground water environment.

O.1.28 Buried Piping and Tanks Inspection Program

Buried Piping and Tanks Inspection Program includes:

(a) Preventive measures such as protective coatings and wrapping are used to mitigate corrosion.

(b) Buried piping will be inspected for coating damage when excavated for any reason. If coating damage is found, then a corrosion inspection will be performed. Before the tenth year of extended operation, BFN will perform an engineering evaluation to determine if sufficient inspections have been conducted to draw a conclusion regarding the ability of the underground coatings to protect the underground piping from degradation. If not, BFN will conduct a focused inspection to allow that conclusion to be reached."

(c) There are no buried tanks within the scope of license renewal.

O.1.29 ASME Section XI, Subsection IWE Program

10 CFR 50.55a imposes the inservice inspection requirements of the ASME B&PV Code Section XI for steel containments (Class MC). The ASME B&PV Code Section XI Subsection IWE Inservice Inspection Program includes visual examination and augmented examinations (visual and/or volumetric examination). Inspections or testing are conducted on the steel containment shells and their integral attachments; containment hatches and airlocks; seals, gaskets, and moisture barriers; and pressure retaining bolting in accordance with ASME B&PV Code Section XI, Subsection IWE, as described in UFSAR Section 4.12, Inservice Inspection and Testing.

The ASME Section XI, Subsection IWE Program requires ultrasonic inspections of the Units 1, 2, and 3 drywell liner plate near the sand bed region. The first inspection on each unit will was completed prior to the period of extended operation. Subsequent periodic inspections will be performed on each unit at a period not to exceed 10 years. The results of these inspections are required to be reviewed to ensure that the acceptance criteria of ASME Section XI, Subsection IWE-3000 are met.

O.1.30 ASME Section XI, Subsection IWF Program

10 CFR 50.55a imposes the inservice inspection (ISI) requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, for Class 1, 2, and 3 piping and components and their associated supports. ISI of supports for ASME piping and components is addressed in Section XI, Subsection IWF. The ASME Section XI, Subsection IWF Program consists of periodic visual examination of ASME Section XI, Class 1, 2, and 3 piping and component supports for signs of degradation, evaluation, and establishment of corrective actions. The Units 1, 2, and 3 programs are in accordance with ASME Section XI, Subsection IWF, as described by Section 4.12, Inservice Inspection and Testing. The IWF program includes ASME Section XI, Class MC equivalent component supports.

O.1.31 10 CFR 50 Appendix J Program

The 10 CFR 50 Appendix J 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 the primary containment pressure boundary. Seals, gaskets, and bolted connections are also monitored. Type A and Type B containment leak rate tests are performed in accordance with the regulations in 10 CFR 50, Appendix J, Option B; and the guidance provided in 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."

O.1.32 Masonry Wall Program

The Masonry Wall program provides for condition monitoring of masonry walls. The program is included in the Structures Monitoring Program that implements the structures monitoring requirements of 10 CFR 50.65. Masonry wall condition monitoring is based on guidance provided in NRC Bulletin 80-11 "Masonry Wall Design" and Information Notice 87-67 "Lessons Learned from Regional Inspections of Licensee Actions in Response to I.E. Bulletin 80-11". Visual inspections are performed consistent with techniques identified in industry codes and standards such as ACI 349.3, "Evaluation of Existing Nuclear Safety-Related Concrete Structures," and ANSI/ASCE 11-90, "Guideline for Structural Condition Assessment of Existing Buildings."

The Masonry Wall Program was enhanced to identify structures with masonry walls within the scope of license renewal, and to clarify inspector qualification requirements to align with industry standards.

O.1.33 Structures Monitoring Program

The Structures Monitoring Program includes periodic inspection and monitoring of the condition of accessible areas of structures. The Structures Monitoring Program implements the requirements of 10 CFR 50.65, the Maintenance Rule that incorporates the guidance of Nuclear Regulatory Commission Regulatory Guide 1.160, "Monitoring and Effectiveness of Maintenance at Nuclear Power Plants," and NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." The Structures Monitoring Program provides inspection guidelines and walkdown checklist for concrete features, roofs, structural steel, masonry walls, seismic gaps, tanks, earthen structures, and miscellaneous components such as doors, suspended systems supports, Non-ASME equivalent pipe supports, and electrical component supports.

The Structures Monitoring Program was enhanced to identify the structures and structural components within the scope of license renewal and to identify the applicable aging effects and mechanisms to be inspected. The program was also enhanced to include examinations of below-grade concrete when excavated for any reason and to clarify inspector qualification requirements to align with industry standards.

O.1.34 Inspection of Water-Control Structures Program

The Inspection of Water-Control Structures program manages age related deterioration, degradation due to extreme environmental conditions, and the effects of natural phenomena that may affect water-control structures. The program is included in the Structures Monitoring Program that implements the structures monitoring requirements of 10 CFR 50.65. The program includes inservice inspection and surveillance activities for dams, slopes, canals, and other water-control structures.

The Inspection of Water-Control Structures Program was enhanced to identify the structures and structural components within the scope of license renewal and to require special inspections following the occurrence of large floods, earthquakes, tornadoes, and intense rainfall. Raw water in close proximity to the Intake Pumping Station is required to be periodically monitored for the requirements of an aggressive environment as described in NUREG-1557.

O.1.35 Environmental Qualification Program

The Environmental Qualification (EQ) Program is imposed by 10 CFR 50.49. The EQ Program manages thermal, radiation, and cyclical aging for components subject to 10 CFR 50.49 requirements through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods. Components subject to 10 CFR 50.49 requirements not qualified for the license term are to be refurbished, replaced, or have their qualification extended prior to reaching the aging limits established in the evaluation.

O.1.36 Fatigue Monitoring Program

The Fatigue Monitoring Program is used for management of metal fatigue of select components in the reactor coolant pressure boundary and primary containment. It provides for monitoring fatigue stress cycles to ensure that the design fatigue usage factor limit is not exceeded.

This program will use EPRI Licensed FatiguePro© Cycle Counting and Fatigue Usage Tracking Computer Program.

O.2 PLANT SPECIFIC AGING MANAGEMENT PROGRAMS

O.2.1 Systems Monitoring Program

The BFN Systems Monitoring Program includes visual inspections of systems and components material conditions. Periodic visual inspections performed during walkdowns identify material condition (i.e., loss of material, corrosion, etc.) prior to the loss of the systems and components intended function, operation, and configuration. Corrective actions are taken to correct deficiencies.

O.2.2 Bus Inspection Program

The Bus Inspection Program provides reasonable assurance that the intended functions of isolated and nonsegregated phase bus will be maintained consistent with the current licensing basis through the period of extended operation. This program manages nonsegregated phase bus insulation exposed to adverse localized environments caused by heat in the presence of oxygen and loosening of fastening hardware associated with isolated and nonsegregated phase bus due to cyclic loading resulting in thermal expansion and contraction of the bus. The program includes inspection of the bus enclosure. The Bus Inspection Program manages the sections of nonsegregated phase bus that connect the Common Station Service Transformers to the 4kV Unit Start Board and the Unit Station Service Transformers to the 4kV Unit Boards. The program also manages the sections of isolated phase bus that connect the Unit Station Service Transformers to the Main Transformers.

O.2.3 Diesel Starting Air Program

The Diesel Starting Air Program includes:

(a) Preventive actions - Filter and membrane dryer replacement minimizes corrosion and corrosion product buildup.

(b) Condition monitoring - Periodic inspections verify the effectiveness of the preventive actions and detect and correct degraded conditions prior to loss of function.

The Diesel Starting Air Program is implemented by the Preventive Maintenance Program. The frequencies for replacements and inspections are established and maintained in accordance with the Preventive Maintenance Program. The diesel starting air piping and receivers were inspected for loss of material under the One-Time Inspection Program.

O.2.4 <u>Unit 1 Periodic Inspection Program</u>

The Unit 1 Periodic Inspection program performs periodic inspections of the non-replaced piping/fittings that were not in service supporting operation of Units 2 and 3 following the extended Unit 1 outage to verify that no latent aging effects are occurring and to connect degraded conditions prior to loss of function.

The piping in the program is carbon/low-alloy or stainless steel that: 1) was exposed to air, treated water or raw water during the extended Unit 1 shutdown; and 2) is exposed to treated water or raw water during normal operation. The inspection locations were selected from non-replaced piping which is in-scope for license renewal and include areas where degradation would be expected as well as areas where degradation would not be expected. The sample selected for periodic inspection was based on a distribution of common material and environment bases, using engineering judgment and design as-built knowledge of the replaced piping in the plant. For a large or infinite lot size, NUREG-1475, “Applying Statistics,” requires a minimum sample size of 59 locations for each material and environment combination. This distribution could not be applied to Unit 1 due to the extensive replacement of candidate piping of one inch diameter or greater.

The initial sample will be utilized in subsequent inspections. The initial baseline inspection of the sample locations was performed prior to restart. The first Unit 1 periodic inspection of all sample locations was performed after Unit 1 return to operation but prior to the end of the current operating period. The second periodic inspection of all sample locations will be completed within the first 10 years of the period of extended operation. The inspection frequency is re-evaluated each time the inspection is performed and can be changed based on the trend of the results. The inspections will continue until the trend of the results provides a basis to discontinue the inspections. However, as a minimum, periodic inspections of all selected sample locations must be performed: 1) after Unit 1 is returned to operation but prior to the end of the current operating period; and 2) within the first ten years of the period of extended operation.

The inspection techniques utilized evaluate internal conditions that are sensitive to the presence of unacceptable conditions including wear, erosion, and corrosion (including crevice corrosion) if present. if unacceptable degradation is detected in any sample location, the unacceptable degradation will be evaluated and dispositioned using the Corrective Action Program.

O.2.5 Carbon Steel/Raw Uncontrolled Water Monitoring Program

The Carbon Steel/Raw Uncontrolled Water Monitoring Program is a condition monitoring program that manages the aging of the equipment and floor drain piping in the Radioactive Waste Treatment System. Periodic inspections are performed to monitor the condition of the piping and ensure that the intended function is maintained.

O.3 TIME-LIMITED AGING ANALYSIS SUMMARIES

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

O.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. The following TLAA discussions are related to the issue of neutron embrittlement:

- Reactor Vessel Materials Upper-Shelf Energy Reduction Due to Neutron Embrittlement
- Adjusted Reference Temperature for Reactor Vessel Materials Due to Neutron Embrittlement
- Reflood Thermal Shock Analysis of the Reactor Vessel
- Reflood Thermal Shock Analysis of the Reactor Vessel Core Shroud
- Reactor Vessel Thermal Limit Analysis: Operating Pressure–Temperature Limits
- Reactor Vessel Circumferential Weld Examination Relief
- Reactor Vessel Axial Weld Failure Probability

O.3.1.1 Reactor Vessel Materials Upper-Shelf Energy Reduction due to Neutron Embrittlement

10 CFR 50 Appendix G requires the predicted end-of-life Charpy impact test upper shelf energy for reactor vessel materials to be at least 50 ft-lb (absorbed energy), unless an approved analysis supports a lower value. The upper shelf energy is the standard industry parameter used to indicate the maximum toughness of a material at high temperature. The 60 year end-of-life upper shelf energy was evaluated for the BFN reactor vessels by using an equivalent margin analysis methodology approved by the NRC in NEDO-32205-A, "10 CFR 50 Appendix C Equivalent Margin Analysis for Low Upper-Shelf Energy in BWR-2 Through BWR-6 Vessels." The results show that the limiting upper shelf energy equivalent margin analysis percent is less than the EPRI Report TR-113596, "BWR Vessel and Internals Project, BWR

Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines (BWRVIP-74)," equivalent margin analysis percent acceptance criterion in all cases, and is therefore acceptable.

O.3.1.2 Adjusted Reference Temperature for Reactor Vessel Materials due to Neutron Embrittlement

10 CFR 50 Appendix G defines the fracture toughness requirements for the life of the reactor pressure vessel. The initial nil-ductility reference temperature (RTNDT) is the temperature at which a non-irradiated metal (ferritic steel) changes in fracture characteristics going from ductile to brittle behavior. An increase (ÄRTNDT) means that higher temperatures are required for the material to continue to act in a ductile manner. The adjusted reference temperature (ART) is defined as RTNDT + ÄRTNDT + margin. The 60 year end-of-life ÄRTNDT for each BFN reactor pressure vessel beltline materials was calculated based on the embrittlement correlation found in Regulatory Guide 1.99. The calculation results show that the limiting 60 year end-of-life ARTs allow pressure-temperature limits that will provide reasonable operational flexibility.

O.3.1.3 Reflood Thermal Shock Analysis of the Reactor Vessel

The design basis for the reactor vessel includes an end-of-life thermal shock analysis performed on the reactor vessels for a design basis LOCA followed by a low pressure coolant injection. The effects of embrittlement assumed by this thermal shock analysis will change with an increase in the licensed operating period. A revised analysis shows that the minimum temperature predicted for the thermal shock event at the time and location of peak stress intensity remains well above the limiting adjusted reference temperature (ART) during the period of extended operation.

O.3.1.4 Reflood Thermal Shock Analysis of the Reactor Vessel Core Shroud

The reactor vessel core shrouds were evaluated for a low pressure coolant injection reflood thermal shock transient considering the embrittlement effects of 40-year radiation exposure (32 EFPY, Effective Full Power Year). The analysis was revised for the 60-year radiation exposure using the approved fluence methodology described in Section O.3.1.2. The results show that the calculated thermal shock strain at the most irradiated location is acceptable considering the embrittlement effects for a 60-year operating period.

O.3.1.5 Reactor Vessel Thermal Limit Analyses: Operating Pressure-Temperature Limits (P-T)

10 CFR Part 50 Appendix G requires reactor vessel thermal limit analyses to determine operating pressure-temperature limits for heatup, cooldown, criticality, and inservice leakage and hydrostatic testing. Because of the relationship between the operating pressure-temperature limits and the fracture toughness transition of the reactor vessel, new operating pressure-temperature limit curves have been calculated and approved for all three units for the extended period of operation.

O.3.1.6 Reactor Vessel Circumferential Weld Examination Relief

Units 2 and 3 have received relief (Reference 1) from reactor vessel circumferential weld examination requirements under Generic Letter 98-05, “Boiling Water Reactor Licensees Use of BWRVIP-05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds," for the remainder of the 40 year licensed operating period. In addition, Unit 1 has received relief (Reference 5) from reactor vessel circumferential weld examination requirements under General Letter 98-05 for the remainder of the 40 year licensed operating period. The circumferential weld examination relief analyses are based on probabilistic assessments that predict an acceptable probability of failure per reactor operating year. The analysis is based on reactor vessel metallurgical conditions as well as flaw indication sizes and frequencies of occurrence that are expected at the end of a licensed operating period.

Although a conditional failure probability has not been recalculated, an analysis that concluded values at the end of a 60 year life are less than the 64 EFPY value provided by the NRC leads to the conclusion that the BFN reactor vessel conditional failure probability is bounded by the NRC analysis in its safety evaluation report (SER) for BWRVIP-05 (Reference 2). The procedures and training used to limit cold over-pressure events will be the same as that approved by the NRC when BFN requested the BWRVIP-05 technical alternative be used for the current term. Relief requests are submitted to the NRC for approval with each unit’s 10-year Interval ISI program.

O.3.1.7 Reactor Vessel Axial Weld Failure Probability

The BWRVIP-05 recommendations for inspection of reactor vessel shell welds contain generic analyses supporting an NRC SER (Reference 3) conclusion that the generic plant axial weld failure rate is no more than 5 x 10-6 per reactor year. BWRVIP-05 showed that this axial weld failure rate of 5 x 10-6 per reactor year is 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 O.3.1.6. The BFN limiting weld chemistry, chemistry factor and 60-year life mean RT_{NDT} values for Units 1, 2, and 3, are within the limits of the values assumed in the analysis performed by the NRC staff in its BWRVIP-05 SER supplement (Reference 4). Therefore, the probability of failure for the axial welds is bounded by the NRC evaluation.

O.3.2 Metal Fatigue

The following thermal and mechanical fatigue analyses of mechanical components have been identified as TLAAs.

- Reactor Vessel Fatigue Analysis
- Reactor Vessel Internals Fatigue Analysis
- Reactor Coolant Pressure Boundary Piping and Component Fatigue Analysis
- Effects of Reactor Coolant Environment on Fatigue Life of Components and Piping (Generic Safety Issue 190)

O.3.2.1 Reactor Vessel Fatigue Analysis

Reactor vessel fatigue analyses of the vessel support skirt, shell, upper and lower heads, closure flanges, nozzles and penetrations, nozzle safe ends, and closure studs, depend on assumed numbers and severity of normal and upset-event pressure and thermal operating cycles to predict end-of-life fatigue usage factors. These assumed cycle counts and fatigue usage factors are based on 40 years of operation. Calculation of fatigue usage factors is part of the current licensing basis and is used to support safety determinations.

The fatigue cumulative usage factors (CUF) for the limiting components have been reevaluated for the period of extended operation. Several components have 60-year CUFs greater than the ASME Code allowable of 1.0.

The Fatigue Monitoring Program (O.1.36) will ensure that fatigue effects will be adequately managed and will be maintained within code design limits for the period of extended operation.

O.3.2.2 Reactor Vessel Internals Fatigue Analysis

Fatigue analysis of the reactor internals was performed using the ASME Boiler and Pressure Vessel Code, Section III, as a guide. The most significant fatigue loading occurs at the jet pump diffuser to baffle plate weld location. Additionally, BFN Unit 3 performed a repair at the core spray T-box to address cracking and replaced a lower section of the core spray line and fatigue analyses were performed using ASME III as a guide.

The jet pump diffuser to baffle plate weld and the T-Box repair fatigue analyses were reevaluated and remain acceptable for a 60 year life. The lower sectional replacement, design life was specified as 40 years. Since the modification was installed more than 20 years into the original license period, the fatigue analysis remains valid for the period of extended operation. The fatigue analyses of the reactor internal components are acceptable for the period of extended operation.

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

The reactor coolant pressure boundary (RCPB) was designed to USAS B31.1-1967. The non-RCPB piping in the scope of license renewal was also designed to USAS B31.1-1967. While this code did not require fatigue analyses, it does require the application of a stress range reduction factor to the allowable stress range for expansion stresses to account for cyclic thermal conditions. A stress range reduction factor of 1.0 is used for the analysis of this piping. Using a stress range reduction factor of 1.0 is applicable when there are 7,000 or less equivalent full-temperature thermal cycles.

The applicability of using a stress range reduction factor of 1.0 for 60 years of plant operation was evaluated. The results of this evaluation indicate that the 7000 thermal cycle assumption is valid and bounding for 60 years of operation. The existing piping analyses within the scope of license renewal containing the assumed thermal cycle counts are valid for the period of extended operation.

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

Generic Safety Issue 190 was identified by the NRC staff because of concerns about the potential effects of reactor water environments on component fatigue life during the period of extended operation. Plant specific evaluations were performed for the locations identified in NUREG/CR-6260, “Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components.” For each of these locations, environmental fatigue calculations were performed using the appropriate Environmental Fatigue Life Correction Factor (F_{en}) relationships for carbon/low alloy steels from NUREG/CR-6583, “Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels,” and stainless steels from NUREG/CR-5704, ”Effects of LWR Coolant Environments on

Fatigue Design Curves of Austenitic Stainless Steels." The locations which have projected CUF values greater than 1.0 will be included in the Fatigue Monitoring Program (O.1.36).

O.3.3 Environmental Qualification of Electrical Equipment

The analyses that establish a qualified life of at least forty years for electrical components subject to the requirements of 10 CFR 50.49 are Time-Limited Aging Analyses as defined by 10 CFR 54.21. The aging effects of electrical components subject to the requirements of 10 CFR 50.49 will be managed in the Environmental Qualification Program in accordance with the requirements of 10 CFR 54.21(c)(1)(iii) for the period of extended operation.

O.3.4 Containment Fatigue

The BFN Torus Integrity Long-Term Program Plant Unique Analysis Report describes the fatigue analyses for the suppression chamber, torus vents, torus downcomers, torus attached pipe, and main steam relief valve discharge lines. The fatigue analyses considered the effects to 500 main steam relief valve actuations during 40 years of normal operation followed by small, intermediate, or design basis pipe break discharges into the suppression pool. Reanalyses were performed to determine the effects of the extended period of operation. Since the 60 year cumulative usage factors were extrapolated to exceed 0.66 for the downcomer/vent header intersection, fatigue will be managed by the Fatigue Monitoring Program (O.1.36). Specifically, main steam safety relief valve lifts will be monitored to ensure that corrective actions are taken before cumulative usage factors approach 1.0.

The design life of the torus vent line and process penetration bellows is 7000 cycles. The number of cycles expected on these bellows has been conservatively approximated by the number of reactor vessel thermal cycles (UFSAR 4.2.5). The expected number of fatigue cycles expected for the 60 year plant life is a small fraction of the design cycles. Fatigue of the torus vent line and process penetration bellows has been dispositioned by confirming that the analyses remain valid for the period of extended operation.

O.3.5 Other Plant Specific Time-Limited Aging Analysis

TVA has evaluated the following to be plant specific TLAAs:

- O.3.5.1 Reactor Building Crane Load Cycles

- O.3.5.2 Radiation Degradation of Drywell Expansion Gap Foam
- O.3.5.3 Irradiation Assisted Stress Corrosion Cracking of Reactor Vessel Internals
- O.3.5.4 Stress Relaxation of the Core Plate Hold-Down Bolts
- O.3.5.5 Emergency Equipment Cooling Water Weld Flaw Evaluation

O.3.5.1 Reactor Building Crane Load Cycles

The reactor building overhead crane is designed to meet the design loading requirements of the Crane Manufacturers Association of America (CMAA) Specification 70, “Specifications for Electric Overhead Traveling Cranes.” For cyclic loading, CMAA 70 specifies that a crane classified as Service Class A1 is limited to 100,000 loading cycles (i.e., 100,000 lifts at rated capacity) over the design life. The 60-year cycle estimate is less than 1,000 cycles at rated capacity and less than 21,000 lift cycles total. The reactor building crane has been evaluated and is qualified for the period of extended operation.

O.3.5.2 Radiation Degradation of Drywell Expansion Gap Foam

The steel drywell shell is enclosed in reinforced concrete for shielding purposes to provide additional resistance to deformation and buckling of the drywell over areas where the concrete backs up the steel shell. Above the transition zone, the drywell is separated from the reinforced concrete by a gap of approximately 2 inches. This gap is filled with polyurethane foam. As described in UFSAR 5.2.3.2, irradiation tests have shown that no change in the resilient characteristics will take place for exposures up to 10^8 Rads. The effect of a postulated increase in the foam stiffness resulting from radiation dose is a TLAA.

Because the analysis shows that predicted exposure will be less than half of the qualified radiation exposure, the material properties of the foam will remain within the limits assumed by the original design analysis, in accordance with the aging assumptions assumed by the original design, for the period of extended operation.

O.3.5.3 Irradiation Assisted Stress Corrosion Cracking of Reactor Vessel Internals

Austenitic stainless steel reactor vessel internal components exposed to a neutron fluence greater than 5×10^{20} n/cm^2 (E > 1 MeV) are considered susceptible to irradiation assisted stress corrosion cracking (IASCC) in the BWR environment. Fluence calculations have been performed for the reactor vessel and internals. Four components have been identified as being susceptible to IASCC for the period of extended operation: (1) Top Guide; (2) Shroud; (3) Core Plate and (4) Incore Instrumentation Dry Tubes and Guide Tubes. IASCC of three components (top guide, shroud and incore instrumentation dry tubes and guide tubes) have been evaluated by the BWRVIP, as described in the Inspection and Evaluation Guidelines for each component: BWRVIP-26 (Top Guide), BWRVIP-76 (Shroud), and BWRVIP-47 (incore instrumentation dry tubes and guide tubes). In addition to the BWRVIP inspections for BWRVIP-26, BWRVIP-47, and BWRVIP-76, the Chemistry Control Program (O.1.5) and the BWR Vessel Internals Program (O.1.12) implement the recommendations of BWRVIP-25 for the core plate.

O.3.5.4 Stress Relaxation of the Core Plate Hold-Down Bolts

The core plate hold-down bolts connect the core plate to the core shroud. These bolts are subject to stress relaxation due to thermal and irradiation effects. For the 40-year lifetime, the BWRVIP concluded that all rim hold-down bolts will maintain some preload throughout the life of the plant. TVA performed a BFN plant-specific analysis consistent with BWRVIP-25 to demonstrate that the core plate hold-down bolts can withstand normal, upset, emergency, and faulted loads, as applicable, considering the effects of stress relaxation until the end of the period of extended operation. The installed core plate configuration and bolt preload was used for the plant specific analysis. The analysis used the plant-specific design basis loads and load combinations. The analysis incorporated detailed flux/fluence analyses and improved stress relaxation correlations.

As per BFN's current licensing basis, UFSAR Section 3.3.5.1, Reactor Vessel Internals Mechanical Design, the ASME Boiler and Pressure Code, Section III was used as a guide in determining limiting stress intensities for reactor vessel internals. For those components for which stresses exceed the ASME code allowables, either the elastic stability of the structure or the resulting deformation of displacement was examined to determine if the safety design basis is satisfied.

The analysis concluded that the BFN core plate bolts meet the ASME allowable stresses for the most limiting plant-specific load combinations and loads for all three scenarios analyzed in BWRVIP-25 Appendix A. The results of the analysis were accepted in an NRC SER (Reference 6).

O.3.5.5 Emergency Equipment Cooling Water Weld Flaw Evaluation

Analysis was performed on a selected number of EECW system piping welds which have flaws that are larger than normally considered acceptable. The analysis included a stress evaluation of the flawed welds and fatigue crack growth calculations. The fatigue crack growth calculations were based on a conservative projection of 125 cycles. Review of the system operation indicated that the projection of 125 cycles remains valid for the period of extended operation.

O.4 10 CFR 54.37(b) NEWLY IDENTIFIED SYSTEMS, STRUCTURES, AND COMPONENTS

10 CFR 54.37(b) states that after a renewed license is issued, the Final Safety Analysis Report (FSAR) update required by 10 CFR 50.71(e) must include any systems, structures, and components (SSCs) newly identified that would have been subject to an aging management review or evaluation of time-limited aging analyses in accordance with 10 CFR 54.21 and the FSAR update must describe how the effects of aging will be managed such that the intended function(s) in 10 CFR 54.4(b) will be effectively maintained during the period of extended operation. The intent of 10 CFR 54.37(b) is to capture those SSCs that, if they had been identified at the time of the license renewal application, would have been subject to an aging management review or evaluation of time-limited aging analyses. This section of Appendix O contains the newly identified SSCs pursuant to the requirements of 10 CFR 54.37(b).

O.4.1 Gate Structure Number 2

Gate Structure Number 2 is a Class I structure within the scope of 10 CFR 54.4(a)(1) because it provides the protective safety function of limiting the potential thermal mixing of the warmer water discharged from the cooling towers during their operation with the water in the intake pumping station channel which provides the makeup for the RHRSW system. Gate Structure Number 2 is common for all three units and is located in the cool water channel between the discharge control structure and the intake channel (forebay). Additional details for Gate Structure Number 2 are found in BFN UFSAR Section 12.2.7.8.4.

The entire structure contains components requiring an aging management review. The component types that require aging management review are piles, sheet piles with crushed rock fill, reinforced concrete walls and slabs (for flow distribution and structural support); and structural steel beam, columns, and plates (for structural support).

The materials of construction for Gate Structure Number 2 components are carbon steel, low alloy steel, and reinforced concrete.

Gate Structure Number 2 components are exposed to buried, embedded/encased, outside air, and submerged environments.

Aging effects requiring management associated with Gate Structure Number 2 are:

- Cracking, loss of bond, loss of material (spalling, scaling) due to corrosion of embedded steel

- Expansion and cracking due to reaction with aggregates
- Increase in porosity and permeability, loss of strength due to leaching of calcium hydroxide or aggressive chemical attack
- Loss of material (spalling, scaling) and cracking due to freeze-thaw
- Loss of material due to general corrosion, crevice corrosion, pitting corrosion

The following aging management programs manage the aging effects for Gate Structure Number 2 components.

- Inspection of Water-Control Structures Program (BFN UFSAR Section O.1.34)

O.4.2 Discharge Control Structure

The Discharge Control Structure is seismically unclassified and is therefore a non-Class I structure within the scope of 10 CFR 54.4(a)(2) because as an element of the Auxiliary Condenser Cooling Water System, it acts as a weir providing flow distribution in the cold water channel flow during periods of cooling tower operations. The height of the concrete portion is 3.5 feet above the bottom of the cold water channel and distributes the cooling tower discharge flow to a point of entering the discharge diffusers via Gate Structure Number 1 during periods of cooling tower operation. This distribution of flow limits the potential overflowing of the safety-related Gate Structure Number 2 such that the ultimate heat sink is protected during periods of cooling tower operation. The structure is common for all three units and is located to the west of Gate Structure Number 2. Additional details for the Discharge Control Structure are found in the BFN UFSAR section 12.2.7.8.

The entire structure contains components requiring an aging management review. The component types that require aging management review are sheet piles with crushed rock fill, and reinforced concrete walls and slabs (for flow distribution and nonsafety-related structural support).

The materials of construction for the Discharge Control Structure components are carbon and low alloy steel (sheet piles), and reinforced concrete.

The Discharge Control Structure components are exposed to buried, outside air, and submerged environments.

Aging effects requiring management associated with Discharge Control Structure are:

- Cracking, loss of bond, loss of material (spalling, scaling) due to corrosion of embedded steel
- Expansion and cracking due to reaction with aggregates
- Increase in porosity and permeability, loss of strength due to leaching of calcium hydroxide or aggressive chemical attack
- Loss of material (spalling, scaling) and cracking due to freeze-thaw
- Loss of material due to general corrosion, crevice corrosion, pitting corrosion
- Loss of material due to abrasion and cavitation

The following aging management programs manage the aging effects for the Control Discharge Structure components.

- Inspection of Water-Control Structures Program (BFN UFSAR Section O.1.34)

O.5 REFERENCES

1. NRC letter to TVA, “Browns Ferry Nuclear Plant Unit 2, Relief Request 2-ISI-9, Alternatives for Examination of Reactor Pressure Vessel Shell Welds (TAC No. MA8424),” 11/18/1999, and NRC letter to TVA, “Browns Ferry Nuclear Plant Unit 3, Relief Request 3-ISI-1, Revision 1, Alternatives for Examination of Reactor Pressure Vessel Shell Welds (TAC No. MA5953),” 11/18/1999

2. NRC letter from Gus C. Lainas to Carl Terry, Niagara Mohawk Power Company, BWRVIP Chairman, “Final Safety Evaluation of the BWRVIP Vessel and Internals Project BWRVIP-05 Report,” (TAC No. M93925), July 28, 1998.

3. NRC letter from Gus C. Lainas to Carl Terry, Niagara Mohawk Power Company, BWRVIP Chairman, “Final Safety Evaluation of the BWRVIP Vessel and Internals Project BWRVIP-05 Report,” (TAC No. M93925), July 28, 1998

4. NRC letter from Jack R. Strosnider, to Carl Terry, BWRVIP Chairman, “Supplement to Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report,” (TAC No. MA3395), March 7, 2000

5. NRC letter to TVA, “Browns Ferry Nuclear Plant Unit 1 - Safety Evaluation for Relief Request 1-ISI-19 Associated with Reactor Pressure Vessel Circumferential Shell Welds (TAC No. MC 3151),” May 31, 2005

6. NRC letter to TVA, “Browns Ferry Nuclear Plant, Units 1, 2, and 3 - Review of Tennessee Valley Authority Commitment Related to Core Plate Bolt Stress Analysis (TAC Nos. ME6615, ME6616, and ME6617),” May 30, 2013