APPENDIX A

USAR SUPPLEMENT

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A1.0 Introduction

The application for a renewed operating license is required by 10 CFR 54.21(d) to include an Updated Safety Analysis Report (USAR) Supplement. The supplement must contain summary descriptions of the programs and activities for managing the effects of aging, and of the evaluations of Time-Limited Aging Analyses (TLAAs) for the period of extended operation. This appendix provides the required supplement for the PINGP USAR.

Section A2.0 of this appendix contains summary descriptions of the programs used to manage the effects of aging during the period of extended operation. Section A3.0 contains descriptions of programs used for management of TLAAs during the period of extended operation. Section A4.0 contains summaries of TLAA evaluations applicable to the period of extended operation. Section A5.0 discusses the incorporation of final License Renewal commitments into the USAR.

Following the issuance of the renewed operating license, the summary descriptions of aging management programs and TLAAs provided in Appendix A, and the final list of License Renewal commitments, will be incorporated into the PINGP USAR as part of a periodic USAR update in accordance with 10 CFR 50.71(e). Other changes to specific sections of the PINGP USAR necessary to reflect a renewed operating license will also be addressed at that time. Following inclusion of this information into the USAR, changes to the descriptions of the aging management programs and activities, if any, will be made in accordance with applicable requirements of 10 CFR 50.71(e) and 10 CFR 50.59.

A2.0 Summary Descriptions of Programs that Manage the Effects of Aging

This section provides summaries of programs and activities credited in the License Renewal Application for managing the effects of aging during the period of extended operation. The Aging Management Programs and activities described herein may not exist as discrete programs at PINGP. In many cases they exist as a compilation of various implementing documents. The program summaries provided should be interpreted as summaries of activities to be performed to manage aging, and not as specific commitments to maintain unique programs with the specific titles and content listed.

The Aging Management Programs and activities in this appendix rely on the Quality Assurance Program for the elements of corrective actions, confirmation process, and administrative controls. The Quality Assurance Program and associated procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of the Quality Assurance Topical Report and 10 CFR 50, Appendix B. The corrective actions and administrative controls for both safety related and non-safety related systems, structures and components are accomplished per the existing Corrective Action

Program and PINGP administrative control program, and are applicable to all Aging Management Programs and activities that will be required during the period of extended operation. The confirmation process is part of the Corrective Action Program and includes reviews to assure that corrective actions are adequate, that they are adequately tracked and reported, and that corrective action effectiveness is reviewed. Any follow-up actions required by the confirmation process are documented in accordance with the Corrective Action Program. The corrective actions, confirmation process, and administrative controls of the Quality Assurance Program are applicable to all Aging Management Programs and activities required during the period of extended operation.

A2.1 10 CFR Part 50, Appendix J Program

The 10 CFR Part 50, Appendix J Program provides for containment system examinations and leakage testing in accordance with 10 CFR 50, Appendix J, Option B. The program incorporates guidance of NRC Regulatory Guide 1.163 and Nuclear Energy Institute NEI 94-01. Containment leak rate tests are performed to assure that leakage through the primary reactor containment, and systems and components penetrating primary containment, do not exceed allowable leakage rate values specified in the Technical Specifications. Periodic surveillance of reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment.

A2.2 Aboveground Steel Tanks Program

The Aboveground Steel Tanks Program ensures the integrity of carbon steel tanks in scope of License Renewal that rest on soil or concrete such that the bottom exterior surface is potentially susceptible to corrosion due to the ingress of water, while being inaccessible for visual inspection. The program provides for visual inspections of tank external surfaces down to their contact with the foundation, including any sealants/caulking at the foundation interfaces. It also provides for ultrasonic bottom thickness measurements from inside the tank to determine if significant thinning is occurring on the inaccessible bottom surface of the tank. External tank surfaces are coated with protective paint or coatings to prevent corrosion.

For insulated outdoor tanks, the inspections cover the exterior surface of the insulation, and specifically look for damage to insulation or its outer covering that could permit water ingress, and for discoloration or other evidence that the insulation has been wetted. If insulation damage or wetting is identified, insulation will be removed at the affected location to permit direct inspection of the external tank surface. In addition, sample sections of insulation near the bottom of each insulated outdoor tank (i.e., locations with the highest potential for wetted insulation) will be removed periodically to permit direct inspection of the tank exterior.

This program will be implemented prior to the period of extended operation.

A2.3 ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program

The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program provides for condition monitoring of ASME Class 1, 2 and 3 pressure-retaining components, their welded integral attachments and bolting. The program is implemented in accordance with the requirements of 10 CFR 50.55a, with specified limitations, modifications and NRC-approved alternatives, and applicable provisions of the ASME Boiler and Pressure Vessel Code, Section XI (ASME Section XI). The program includes periodic visual, surface, and/or volumetric examinations, and leakage tests. The program also provides component repair and replacement requirements in accordance with ASME Section XI.

The provisions of ASME Section XI are augmented by additional inspections to detect general and pitting corrosion on the shell to transition cone weld of the Westinghouse Model 51 steam generators in Unit 2. Westinghouse Model 51 steam generators have a high stress region at the shell to transition cone weld, and corrosion of the steam generator shell is known to exist.

The program is updated periodically as required by 10 CFR 50.55a.

A2.4 ASME Section XI, Subsection IWE Program

The ASME Section XI, Subsection IWE Program provides for condition monitoring of Class MC pressure-retaining components and their related items, including integral attachments, seals, gaskets, moisture barriers, and pressure-retaining bolting. The program is implemented in accordance with the requirements of 10 CFR 50.55a, with specified limitations, modifications and NRC-approved alternatives, and applicable provisions of the ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWE. The program monitors for aging effects by performing visual examinations of the Class MC components and their related items. Visual or volumetric examinations, as applicable, are performed on components that require augmented examination.

The program is updated periodically as required by 10 CFR 50.55a.

A2.5 ASME Section XI, Subsection IWF Program

The ASME Section XI, Subsection IWF Program provides for condition monitoring of Class 1, 2 and 3 component supports. The program is implemented in accordance with the requirements of 10 CFR 50.55a, with specified limitations, modifications and NRC-approved alternatives, and the applicable provisions of the ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWF. The program manages aging effects by performing periodic visual examinations of supports for Class 1, 2, and 3 piping and components.

The program is updated periodically as required by 10 CFR 50.55a.

A2.6 Bolting Integrity Program

The Bolting Integrity Program manages the aging affects associated with closure bolting in mechanical components and with structural bolting in the scope of License Renewal through periodic inspection, material selection, thread lubricant control, assembly and torque requirements, and repair and replacement requirements. Inspections of bolting within the scope of the Bolting Integrity Program are conducted under the following programs:

- ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program,
- ASME Section XI, Subsection IWE Program,
- ASME Section XI, Subsection IWF Program,
- Buried Piping and Tanks Inspection Program,
- External Surfaces Monitoring Program,
- RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants Program, and
- Structures Monitoring Program.

A2.7 Boric Acid Corrosion Program

The Boric Acid Corrosion Program is a condition monitoring program developed in accordance with NRC Generic Letter 88-05, “Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants.” The program performs periodic visual examinations of the reactor coolant pressure boundary and other systems containing borated water for evidence of leakage and corrosion. Adjacent structures, components (including electrical), and supports are also examined for boric acid accumulation and corrosion. The program includes evaluations, assessments, and corrective actions for the observed leakage sources and any affected structures and components.

A2.8 Buried Piping and Tanks Inspection Program

The Buried Piping and Tanks Inspection Program manages loss of material on the external surfaces of carbon steel and cast iron components that are buried in soil or sand. As a preventive measure, buried pipe is coated and wrapped prior to initial installation in accordance with standard industry practices to prevent/mitigate corrosion. The program performs visual inspections following excavation of external surfaces of buried components (e.g., piping, tanks, bolting) for evidence of coating damage and degradation of the underlying carbon steel and cast iron. If no evidence of damage to the coating or wrapping is detected, then the coating or wrapping will not be removed for further inspection. The periodicity of these inspections will be based on opportunities for inspection such as scheduled maintenance work, with at least one inspection occurring within ten years prior to the period of extended operation, and one in each ten-year period thereafter. If an opportunity for inspection does not occur within a ten-year period, then a focused inspection of a sample component will be performed prior to the end of that period.

This program will be implemented prior to the period of extended operation.

A2.9 Closed-Cycle Cooling Water System Program

The Closed-Cycle Cooling Water System Program is both a preventive and condition monitoring program that is based on the Electric Power Research Institute (EPRI) closed cooling water chemistry guidelines. The program includes preventive measures (maintenance of system corrosion inhibitor concentrations) to minimize corrosion, heat transfer degradation, and stress corrosion cracking; and testing and inspection to monitor the effects of corrosion, heat transfer degradation, and stress corrosion cracking on the intended functions of the components. In addition, cleaning and inspection of heat exchangers are performed periodically along with pump and heat exchanger performance/functional testing.

A2.10 Compressed Air Monitoring Program

The Compressed Air Monitoring Program is a condition monitoring program that manages the effects of corrosion and the presence of unacceptable levels of contaminants for the Station and Instrument Air System. The program conducts periodic air quality sampling, inspections, component functional testing, and leakage testing. Additionally, preventive maintenance is performed at regular intervals to assure system components continue to operate reliably, thereby assuring that quality air is supplied to plant equipment. This program implements the PINGP commitments made in response to NRC Generic Letter 88-14, "Instrument Air Supply System Problems Affecting Safety-Related Equipment."

A2.11 Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program

The Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program conducts a one-time test of a representative sample of electrical cable connections (metallic portions) to confirm the absence of aging effects (loose connections). Cable connections terminating within an active or passive device/enclosure from external sources are within the scope of this program. Cable/wiring connections terminating within an active or passive device/enclosure from internal sources are not within the scope of this program. The representative sample includes connections of various voltage applications (medium and low voltage), circuit loadings and locations (high temperature, high humidity, vibration, etc.).

This program will be completed prior to the period of extended operation.

A2.12 Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program manages the aging effect reduced insulation resistance on insulated electrical cables and connections (including splices, terminations, fuse blocks, connectors, and insulation portions of electrical penetrations) installed in adverse localized environments (e.g., high temperature, radiation and/or moisture levels significantly more severe than design service conditions) to ensure cable and connection insulation integrity is maintained throughout the period of extended operation. The program conducts periodic visual inspections on a representative sample of accessible cables and connections in identified adverse localized environments, to confirm insulation integrity. Inspections are performed at least once every ten years, with the first inspection completed before the period of extended operation.

This program will be implemented prior to the period of extended operation.

A2.13 Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program

The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program manages the aging effect of reduced insulation resistance on non-EQ, sensitive (high voltage, low signal) instrumentation circuit cables and connections, that are exposed to adverse ambient or adverse localized environments, to maintain electrical circuit integrity. An adverse localized environment is a condition of high temperature, radiation and/or moisture that is significantly more severe than the specified service environment for the cable. This program includes either periodic review of surveillance data, or testing of cables and connections, for high-range-radiation and neutron flux monitoring instrumentation that is sensitive to a reduction in cable insulation resistance. The first reviews/tests are completed before the period of extended operation, and are conducted at least once every ten years thereafter.

This program will be implemented prior to the period of extended operation.

A2.14 External Surfaces Monitoring Program

The External Surfaces Monitoring Program is a condition monitoring program that implements inspections and walkdowns of systems and components within the scope of the program. Periodic system inspections and walkdowns are conducted to visually inspect accessible external surfaces of piping, piping components, ducting, and other metallic and non-metallic components for aging degradation. The program is also credited with managing aging effects of internal surfaces for situations in which the external surface is subject to the same environment or stressor as the internal surface, such that the external surface condition is representative of internal surface condition.

A2.15 Fire Protection Program

The Fire Protection Program is a condition monitoring program which consists of fire barrier inspection activities, diesel-driven fire pump inspection activities and halon/carbon dioxide (CO_2) fire suppression system inspection activities. The fire barrier inspection activities include periodic visual inspection of fire barrier penetration seals, fire barrier walls, ceilings, and floors, and periodic inspection and functional testing of all fire-rated doors that perform a fire barrier function to ensure that their operability and intended functions are maintained. The diesel-driven fire pump inspection activities include periodic pump performance testing to ensure that the fuel supply line can perform its intended function. The halon/CO_2 fire suppression system inspection activities include both periodic inspection and functional testing of the halon/CO_2 fire suppression system to manage the aging effects and degradation that may affect the intended function and performance of the system.

A2.16 Fire Water System Program

The Fire Water System Program is a condition monitoring program that conducts inspections and performance tests of water-based fire protection system components such as sprinklers, nozzles, fittings, valves, hydrants (including hose and gaskets), hose stations, standpipes, and aboveground and underground piping and components. Inspection and testing are performed in accordance with applicable National Fire Protection Association (NFPA) codes and standards, and NRC commitments. Fire protection system piping is subject to periodic flushing and wall thickness evaluations to ensure that corrosion, microbiologically-influenced corrosion (MIC), and fouling are managed such that the system function is maintained. Additionally, internal portions of the fire water system are visually inspected when disassembled for maintenance. Prior to exceeding the 50-year service life, sprinkler heads will be replaced or be subject to representative sample testing.

A2.17 Flow-Accelerated Corrosion Program

The Flow-Accelerated Corrosion (FAC) Program is a condition monitoring program based on Electric Power Research Institute (EPRI) guidelines for an effective FAC program. The program manages loss of material due to FAC in piping and components containing high-energy single phase or two phase fluids. The program includes (a) conducting an analysis to determine critical locations, (b) performing baseline inspections to determine the extent of thinning at these locations, and (c) performing follow-up inspections to confirm predictions of the rate of thinning, or repairing or replacing components as necessary. This program implements the PINGP response to NRC Generic Letter 89-08.

A2.18 Flux Thimble Tube Inspection Program

The Flux Thimble Tube Inspection Program is a condition monitoring program that manages loss of material due to wear for in-core instrument thimble tubes. The program implements periodic eddy current testing of thimble tubes for thinning of the flux thimble tube wall due to flow-induced fretting. The program also provides for evaluation and trending of inspection results and appropriate corrective actions. This program implements the PINGP commitments made in response to NRC Bulletin 88-09, “Thimble Tube Thinning in Westinghouse Reactors.”

A2.19 Fuel Oil Chemistry Program

The Fuel Oil Chemistry Program manages the aging effects of loss of material and cracking on internal surfaces of the diesel fuel oil system piping, piping components and tanks by minimizing the potential for a corrosive environment, and by verifying that the actions taken to mitigate corrosion are effective. The program includes: (1) periodic sampling and testing of stored fuel oil and testing of new fuel oil in accordance with plant Technical Specifications and selected industry standards to confirm water, sediment and contaminants remain below limits of concern for corrosion to occur; (2) periodic testing of fuel oil storage tanks for the presence of water; (3) periodic integrity testing of underground storage tanks and external visual inspections of aboveground storage tanks to confirm leakage is not occurring; and, (4) one-time inspections of selected tank bottom and piping locations, using ultrasonic testing, to be performed prior to the period of extended operation.

A2.20 Fuse Holders Program

The Fuse Holders Program is a condition monitoring program that implements periodic visual inspections and tests of fuse holders in scope of License Renewal, located in passive enclosures and assemblies, and exposed to adverse localized environments. A localized environment is adverse if it promotes loose connections from clip relaxation/fatigue (i.e., ohmic heating, thermal cycling or electrical transients, mechanical fatigue caused by frequent removal/replacement of the fuse, or vibration), or if it exposes the fuse holder to adverse levels of chemical contamination or moisture that would promote corrosion and oxidation of the metallic fuse clips.

Fuse holders determined to be operating in an adverse localized environment will be visually inspected and tested at least once every 10 years, with the first inspections and tests completed before the period of extended operation. The specific type of test performed will be capable of detecting deterioration of metallic clamps of the fuse holders, such as thermography, contact resistance testing, or other appropriate test method.

This program will be implemented prior to the period of extended operation.

A2.21 Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program

The Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program performs periodic tests to provide an indication of the condition of the conductor insulation for medium voltage cables in scope of License Renewal exposed to long periods of high moisture (greater than a few days at a time) and subjected to voltage stress (energized greater than 25 percent of the time). This program includes underground cables (direct buried or in underground ducts) not designed for wet environments. Insulation testing for the affected cables is performed at least once every 10 years, with the first tests completed prior to the period of extended operation.

The program also includes periodic inspections of the applicable underground raceway manhole for the accumulation of water and draining water, if necessary. Manhole inspections are performed at least once every two years, with the first inspection completed before the period of extended operation.

This program will be implemented prior to the period of extended operation.

A2.22 Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program

The Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program is a condition monitoring program that performs visual inspections of the internal surfaces of mechanical components within the scope of License Renewal not covered by other aging management programs. The internal inspections are performed during scheduled preventive and corrective maintenance activities, or during other routinely scheduled tasks such as surveillance procedures, when internal surfaces are made accessible for inspections. The program inspections are performed to provide assurance that existing environmental conditions are not resulting in degradation that could result in a loss of component intended functions.

This program will be implemented prior to the period of extended operation.

A2.23 Inspection of Overhead Heavy Load and LIght Load (Related to Refueling) Handling Systems Program

The Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems Program implements condition monitoring activities that ensure structural components of heavy load handling systems and light load handling systems related to refueling within the scope of License Renewal are capable of sustaining their rated loads for the period of extended operation. The load handling components in scope of License Renewal are overhead heavy load handling components subject to the requirements of NUREG-0612, and light load handling components associated with refueling activities. The program provides for periodic visual inspections of structural components, including crane rails, structural girders, beams, special lifting devices, and welded and bolted connections.

A2.24 Lubricating Oil Analysis Program

The Lubricating Oil Analysis Program obtains and analyzes lubricating and hydraulic oil samples from plant equipment to ensure that oil quality is maintained within acceptable limits to preserve an operating environment that is not conducive to loss of material, cracking, or heat transfer degradation. Program activities include periodic oil sampling, analysis, and evaluation and trending of results.

A2.25 Masonry Wall Program

The Masonry Wall Program is a condition monitoring program that performs periodic visual inspections of masonry walls in proximity to, or with attachments to, safety related equipment. The program is based on guidance provided in NRC IE Bulletin 80-11, "Masonry Wall Design," and NRC Information Notice 87-67, "Lessons Learned from Regional Inspections of Licensee Actions in Response to IE Bulletin 80-11." The Masonry Wall Program assures that the evaluation basis established for each masonry wall within the scope of License Renewal remains valid.

A2.26 Metal-Enclosed Bus Program

The Metal-Enclosed Bus Program is a condition monitoring program that inspects representative samples of the interiors of non-segregated 4160V phase bus between station offsite source auxiliary transformers and plant buses. Internal visual inspection is performed to observe signs of aging to the bus insulation materials (such as cracking and discoloration), evidence of loose connections, and signs of moisture and debris intrusion. Internal bus supports are visually inspected for structural integrity and signs of cracks. The inspection may include thermography and/or electrical resistance testing to ensure the integrity of bus connections. The interior visual inspection is conducted at least once every five years, or, if conducted with thermography or electrical resistance testing, at least once every ten years. The first inspections and/or tests are completed before the period of extended operation.

This program will be implemented prior to the period of extended operation.

A2.27 Nickel-Alloy Nozzles and Penetrations Program

For the Nickel-Alloy Nozzles and Penetrations Program, PINGP commits to the following activities for managing the aging of nickel-alloy components susceptible to primary water stress corrosion cracking:

1. comply with applicable NRC orders, and

2. implement applicable NRC Bulletins, Generic Letters, and staff-accepted industry guidelines.

A2.28 Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors Program

The Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors Program is a condition monitoring program that implements the requirements of the NRC First Revised Order EA-03-009, "Issue of Order Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors," dated February 20, 2004 (Order). This program manages the aging effects of cracking due to primary water stress corrosion cracking of the nickel-alloy vessel head penetration nozzles welded to the upper reactor vessel head. In addition, the program monitors the upper reactor vessel head surface and the region above the reactor vessel head for boric acid leakage.

This program is a mandated augmented inservice inspection program that supplements the leakage tests and visual VT-2 examinations required by ASME Section XI, Table IWB-2500-1, Examination Category B-P. The program incorporates the susceptibility ranking of the upper vessel head penetration nozzles to primary water stress corrosion cracking, and the required process for establishing the inspection methods and inspection frequencies in accordance with the susceptibility ranking, as required by the Order, as amended.

A2.29 One-Time Inspection Program

The One-Time Inspection Program provides additional assurance, through sampling inspections using nondestructive examination (NDE) techniques, that aging is not occurring or that the rate of degradation is so insignificant that additional aging management actions are not warranted. The program includes measures to verify the effectiveness of other aging management programs, such as the Water Chemistry Program, to mitigate aging effects. In other cases, this program confirms that a separate aging management program is not warranted when significant aging is not expected to occur. If aging effects are identified that could adversely impact an intended function prior to the end of the period of extended operation, additional actions will be taken to correct the condition, perform additional inspections, and/or perform periodic inspections as needed.

The program elements 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 inspection locations in the system, component, or structure based on the aging effect; (c) determination of the examination technique, including acceptance criteria that would be effective in managing the aging effect that is being examined; and (d) evaluation of the need for follow-up examination if degradation is identified that could jeopardize an intended function prior to the end of the period of extended operation. The program relies on the results of inspections performed within the 10-year period preceding the period of extended operation.

This program will be completed prior to the period of extended operation.

A2.30 One-Time Inspection of ASME Code Class 1 Small-Bore Piping Program

The One-Time Inspection of ASME Code Class 1 Small-Bore Piping Program is a condition monitoring program that provides additional assurance that aging of Class 1 small-bore piping either is not occurring or is insignificant, such that a new plant-specific aging management program is not warranted. The program inspects for the presence of cracking by performing one-time volumetric examinations on a sample of butt welds in Class 1 piping (including pipes, fittings, and branch connections) less than 4-inch nominal pipe size. The one-time inspections are performed at locations that are determined to be potentially susceptible to cracking based upon the methodology of the site-specific, NRC-approved, Risk Informed Inservice Inspection Program.

This program will be completed prior to the period of extended operation.

A2.31 Open-Cycle Cooling Water System Program

The Open-Cycle Cooling Water (OCCW) System Program implements the commitments made in the PINGP response to NRC Generic Letter 89-13, “Service Water System Problems Affecting Safety-Related Equipment,” to ensure that the effects of aging in OCCW systems, and in components serviced by the OCCW systems, will be managed for the period of extended operation. This program manages aging effects associated with metallic components exposed to a raw water environment. These aging effects are due to corrosion, erosion, and fouling (including silting and coating failure). The program includes (a) surveillance and control of fouling, (b) tests to verify heat transfer capabilities, and (c) routine inspection and maintenance activities.

A2.32 PWR Vessel Internals Program

For the PWR Vessel Internals Program, PINGP commits to the following activities for managing the aging of reactor vessel internals components:

1. participate in the industry programs for investigating and managing aging effects on reactor internals;
2. evaluate and implement the results of the industry programs as applicable to the reactor internals; and
3. upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.

A2.33 Reactor Head Closure Studs Program

The Reactor Head Closure Studs Program implements inservice inspection of reactor vessel head closure studs. The program is implemented in accordance with the requirements of 10 CFR 50.55a, with specified limitations, modifications, and NRC-approved alternatives, and the applicable requirements of the ASME Boiler and Pressure Vessel Code, Section XI. The program includes preventive measures to mitigate cracking including proper material selection, avoiding the use of metal-plated stud bolting, and controlling the use of surface treatments and lubricants.

This program is updated periodically as required by 10 CFR 50.55a.

A2.34 Reactor Vessel Surveillance Program

The Reactor Vessel Surveillance Program manages the reduction of fracture toughness due to neutron embrittlement of the low alloy steel reactor vessels. The program ensures that reactor vessel materials meet the requirements of 10 CFR 50.60 for fracture prevention and 10 CFR 50.61 for Pressurized Thermal Shock (PTS). This program includes surveillance capsule removal and specimen mechanical testing/evaluation, radiation analysis, development of pressure-temperature operating limits, and determination of low-temperature overpressure protection (LTOP) setpoints. Withdrawn untested capsules placed in storage are maintained for future insertion. Monitoring methods are in accordance with 10 CFR 50, Appendix H. Fracture toughness is in accordance with 10 CFR 50, Appendix G. In addition, the program complies with Regulatory Guide 1.99 and ASTM E-185.

The Reactor Vessel Surveillance Program manages updates of pressure-temperature operating limitations and the surveillance specimen withdrawal schedule, as needed, consistent with plant Technical Specifications, the Pressure and Temperature Limits Report, and 10 CFR 50.60 and 10 CFR 50, Appendix H.

A2.35 RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants Program

The RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants Program manages aging effects in water-control structures and components, including bolting, through periodic visual inspections and hydrographic surveys. Program elements include guidance on inspection scope, aids to facilitate the inspection process, criteria used to evaluate the inspection results, guidance on inspection frequency, and documentation requirements. Structures included within the scope of the program are the Screenhouse, Emergency Cooling Water Intake (crib), Intake Canal, and Approach Canal.

This program does not constitute a commitment to the guidance of NRC Regulatory Guide 1.127. RG 1.127 focuses on dams, reservoirs behind those dams, and dam safety and outlet works that deliver cooling water from reservoirs and spill excess water to prevent dam overtopping. These components are not within the scope of License Renewal at PINGP. However, this program considers the guidance in NRC RG 1.127 and ACI 349.3R-96 if it is necessary to evaluate degradation mechanisms and questionable concrete conditions.

A2.36 Selective Leaching of Materials Program

The Selective Leaching of Materials Program performs a one-time visual inspection in conjunction with a hardness measurement, or other suitable detection technique, of selected components in scope of License Renewal made of cast iron, copper alloys >15% zinc, and copper-nickel in environments conducive to selective leaching. Through inspections of representative samples, the program will determine if selective leaching is occurring and, if found, whether the aging mechanism will affect the ability of the component to perform its intended function.

This program will be completed prior to the period of extended operation.

A2.37 Steam Generator Tube Integrity Program

The Steam Generator Tube Integrity Program consists of activities that manage the aging effects cracking, denting, ligament cracking, and loss of material for steam generator tubes, tube plugs, tube repairs and various secondary side internal components. The Steam Generator Tube Integrity Program is implemented in accordance with Technical Specifications Section 5.5.8 and applicable industry guidance. The program manages aging effects through a balance of prevention, inspection, evaluation, repair, and leakage monitoring. Eddy current testing is used to detect steam generator tube flaws and degradation. Visual examinations are conducted on tube plugs, sleeve plugs, and sleeves as necessary. In addition, visual inspections are performed to identify degradation of secondary side steam generator internal components.

A2.38 Structures Monitoring Program

The Structures Monitoring Program is a condition monitoring program that manages aging effects in structures, supports and structural components, including bolting, within the scope of License Renewal. The program performs periodic visual inspections to monitor the condition of structures, supports and components, including bolting, against established acceptance criteria to ensure that degradation is identified, evaluated, and, when necessary, corrected such that there is no loss of intended function.

A2.39 Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program

The Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program manages loss of fracture toughness due to thermal aging embrittlement of CASS components, other than pump casings and valve bodies, that are exposed to reactor coolant operating temperatures. The program determines the susceptibility of CASS components to loss of fracture toughness due to thermal aging embrittlement based on the casting method, molybdenum content, and percent ferrite. For components determined to be potentially susceptible to thermal aging embrittlement, the program provides for enhanced volumetric examinations or component-specific flaw tolerance evaluations. The program augments the PINGP ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD Program.

This program will be implemented prior to the period of extended operation.

A2.40 Water Chemistry Program

The Water Chemistry Program manages aging effects by controlling the internal environment of systems and components. The program mitigates corrosion, stress corrosion cracking and heat transfer degradation due to fouling in the primary, auxiliary (borated), and secondary water systems included in the scope of the program. Aging effects are managed by controlling concentrations of known detrimental chemical species such as chlorides, fluorides, sulfates and dissolved oxygen below the levels known to cause degradation. The program includes specifications for chemical species, sampling and analysis frequencies, and corrective actions for control of water chemistry. This program implements the EPRI PWR primary and secondary water chemistry guidelines.

A3.0 Summary Descriptions of Time-Limited Aging Analyses Aging Management Programs

A3.1 Environmental Qualification (EQ) of Electrical Components Program

The Environmental Qualification (EQ) of Electrical Components Program (EQ Program) implements the requirements of 10 CFR 50.49 (as further defined and clarified by the DOR Guidelines and NUREG-0588), and the guidance of Regulatory Guide 1.89, “Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Plants,” Revision 1. The EQ Program manages component thermal, radiation, and cyclical aging through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods to assure that certain electrical components located in harsh plant environments are qualified to perform their safety functions in those harsh environments. As required by 10 CFR 50.49, EQ components not qualified for the license term are to be refurbished or replaced, or have their qualification extended, prior to reaching the aging limits established in the evaluation.

A3.2 Metal Fatigue of Reactor Coolant Pressure Boundary Program

The Metal Fatigue of Reactor Coolant Pressure Boundary Program monitors the thermal and pressure transients experienced by selected reactor coolant system pressure boundary components to ensure those components remain within their design fatigue usage limits. The program uses the systematic counting of plant transient cycles to ensure that design assumptions for cumulative transient cycles are not exceeded. The program also uses computerized cycle-based or stress-based monitoring methods to track fatigue usage in critical high-usage components. Locations monitored by the program include the six component locations for older vintage Westinghouse plants identified in NUREG/CR-6260 as representative locations for the effect of reactor coolant environment on component fatigue life.

The program ensures that cumulative fatigue usage of each affected primary system location is evaluated, and corrective actions taken if necessary, when the number or magnitude of accumulated thermal and pressure transients approach or exceed design cycle assumptions, or when the projected fatigue usage approaches a value of 1.0, during the life of the plant including the period of extended operation.

A4.0 Summary Descriptions of Evaluations of Time-Limited Aging Analyses

In accordance with 10 CFR 54.21(c), an application for a renewed operating license requires an evaluation of Time-Limited Aging Analyses (TLAAs) for the period of extended operation. The following TLAAs were identified and evaluated to meet this requirement. A summary of the results of each evaluation is provided for each TLAA. These summaries will be incorporated into appropriate locations in the USAR.

A4.1 Reactor Vessel Neutron Embrittlement

The PINGP analyses that address the effects of neutron irradiation embrittlement of the reactor vessels are TLAAs for License Renewal. The analyses have been updated to address twenty additional years of operation during the period of extended operation. For the purpose of projecting fluence and evaluating reactor vessel fracture toughness at 60 years, 54 EFPY is assumed to be the number of effective full power years of operation at the end of the period of extended operation.

Reactor Vessel Fluence

The neutron fluence experienced by critical vessel locations has been projected to the end of the period of extended operation using NRC-approved methodology. The fluence projections were based on operational data through Cycle 24 for Unit 1 and Cycle 23 for Unit 2 at the licensed power level of 1650 MWt. The projections also accounted for a planned thermal power level increase from the Measurement Uncertainty Recapture - Power Uprate (MUR-PU) during Cycle 25. The peak fluence at the clad/base metal interface at 54 EFPY is 5.162E19 n/cm^2 for Unit 1 and 5.196E19 n/cm^2 for Unit 2.

Charpy Upper-Shelf Energy

Appendix G of 10 CFR 50 requires that reactor vessel beltline materials “... must maintain Charpy upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb….”

Fluence values for 54 EFPY at the ¼T location were obtained by applying Equation (3) of Regulatory Guide 1.99 based on a vessel thickness of 6.692 inches. Upper-shelf energies for beltline forgings and welds at 54 EFPY for PINGP Units 1 and 2 are all projected to be above 50 ft-lb.

Pressurized Thermal Shock

10 CFR 50.61(b)(1) provides rules for the protection of pressurized water reactors against pressurized thermal shock. Licensees are required to assess the projected values of reference temperature whenever a significant change occurs in the projected values of the reference temperature for pressurized thermal shock (RT_{PTS}), or upon request for a change in the expiration date for the facility operating license. For License Renewal, RT_{PTS} values were calculated for the projected fluence values at 54 effective full power years (EFPY).

10 CFR 50.61(b)(2) establishes screening criteria for RT_{PTS} of 270^{o}F for plates, forgings, and axial welds, and 300^{o}F for circumferential welds. The values of RT_{PTS} at 54 EFPY for PINGP Units 1 and 2 are all within the established screening criteria. The limiting beltline material for PINGP Unit 1 is the nozzle shell forging B to intermediate shell forging C circumferential weld 2269, with an RT_{PTS} of 157^{o}F at 54 EFPY. The limiting beltline material for PINGP Unit 2 is the nozzle shell forging B to intermediate shell forging C circumferential weld 1752, with an RT_{PTS} of 136^{o}F at 54 EFPY.

Pressure-Temperature Limits

10 CFR 50, Appendix G requires reactor pressure vessel (RPV) thermal limit analyses to determine operating pressure-temperature (P-T) limits for boltup, hydrotest, pressure tests, and normal operating and anticipated operational occurrences. P-T limit curves are developed to satisfy the requirements of 10 CFR 50, Appendix G. Irradiation embrittlement effects are included in the core beltline P-T curve limits.

The PINGP Pressure and Temperature Limits Report contains the P-T limit curves. The P-T limit curves will be updated by the Reactor Vessel Surveillance Program, when required, in accordance with Appendix G of 10 CFR 50.

Low-Temperature Overpressure Protection Analyses

Each time the P-T limit curves are revised, the Low-Temperature Overpressure Protection System (OPPS) limits must be re-evaluated to ensure its functional requirements continue to be met. Calculation of new low-temperature overpressure protection limits is performed by the Reactor Vessel Surveillance Program as part of the development of the pressure-temperature limit curves.

A4.2 Metal Fatigue

Fatigue is an age-related degradation mechanism caused by cyclic stressing of a component by either mechanical or thermal stresses. Fatigue analyses for Class 1 and selected non-Class 1 mechanical components are TLAAs for License Renewal if they meet all six elements of the definition in 10 CFR 54.3(a). Analyses that are based on a number of cycles estimated for the original 40-year license term were considered to have met criterion 54.3(a)(3).

The fatigue evaluations reported in this section are based on normal, upset, and test design transients defined in component design specifications and the USAR. Design basis analyses, where available, were used with the identified aging management program(s), to provide assurance that components will remain within their fatigue usage limits (cumulative usage factor less than 1.0) through the period of extended operation. The design transients and analyses results were also reviewed to assess the impact of the planned Measurement Uncertainty Recapture-Power Uprate (MUR-PU). The review concluded that the impact of the planned MUR-PU on fatigue usage would be very small, and implementation of the MUR-PU in itself would not result in any component reaching a fatigue usage limit during the period of extended operation, or requiring aging management strategies beyond those already discussed.

For purposes of the License Renewal fatigue evaluations, the PINGP Class 1 boundary includes components within the ASME Section XI, Subsection IWB inspection boundary and the steam generator items designed to ASME Section III, Class A or Class 1.

Class 1 and non-Class 1 components determined to be potentially susceptible to fatigue damage and fatigue flaw growth were reviewed for TLAAs and evaluated where applicable. The metal fatigue TLAA evaluation results for Class 1 and non-Class 1 components are summarized below.

<u>Class 1 Metal Fatigue</u>

Class 1 components evaluated for fatigue include the reactor pressure vessels, reactor vessel internals, pressurizers, steam generators, reactor coolant pumps, control rod drive mechanism housings, and Class 1 piping and in-line components. The fatigue analyses calculate a cumulative usage factor (CUF) for a selected component or subassembly based on a specified number of design transient cycles for that component. Design transient cycle assumptions for PINGP Class 1 components are listed in USAR Section 4.1.4 and Table 4.1-8. For the License Renewal evaluations, the numbers of design transient cycles accumulated through September 30, 2006 were projected forward to determine the numbers of cycles expected at the end of 60 years of operation. The numbers of design transient cycles projected to be accumulated at 60 years were less than the numbers of cycles accounted for in the design fatigue analyses for 40 years. Therefore, the original number of design transient cycles will remain valid through the period of extended operation. As a result, with the exception of the reactor vessel internals baffle bolts, the design fatigue analyses of Class 1 components based on those transients will remain valid for the period of extended operation.

In the case of the Reactor Vessel Internals, the fatigue assessment concluded that the limiting items in the baffle plate assembly, the baffle bolts, are not capable of sustaining the full set of plant loading at 5% per minute and plant unloading at 5% per minute design cycles. The total number of allowable cycles of the plant loading and unloading design transient were reassessed to determine a reduced number of cycles that would limit the total baffle bolt CUF to less than 1.0. The total allowable number of cycles was determined to be 1835 compared to the original design value of 18,300. The relevant transients, plant loading and unloading at 5% per minute, are projected to occur only 970 times over the 60-year operating period, well below the reduced cycle limit for the baffle bolts of 1835. USAR Table 4.1-8 is being revised to impose this additional cyclic limit for baffle bolt fatigue. With this reduced cyclic limit, the TLAA for baffle bolts has been projected through the period of extended operation.

<u>Non-Class 1 Metal Fatigue</u>

Non-Class 1 mechanical components that are within the scope of License Renewal and subject to fatigue evaluation fell into two major categories: (1) piping and in-line components (tubing, piping, traps, thermowells, valve bodies, etc.), or (2) non-piping components (tanks, vessels, heat exchangers, pump casings, turbine casings, etc.).

For non-Class 1 piping and in-line components identified as potentially susceptible to cracking due to fatigue, a review of system operating characteristics was conducted to determine the approximate frequency of any significant thermal cycling. If the number of equivalent full temperature cycles experienced in 60 years is below the limit used for the original design (typically 7000 cycles for a stress range reduction factor of 1.0), the component fatigue life is suitable for extended operation. If the number of equivalent full temperature cycles exceed the limit, the individual stress calculations require evaluation. No PINGP systems were projected to exceed 7000 full temperature cycles at 60 years. Therefore, the TLAAs for non-Class 1 piping and in-line components remain valid for the period of extended operation.

The only non-Class 1, non-piping components identified with fatigue-related TLAAs were the auxiliary heat exchangers (sample heat exchangers, residual heat exchangers, regenerative heat exchangers, letdown heat exchangers, and excess letdown heat exchangers). The design transients identified in the equipment specifications were determined to be consistent with the design transients defined for 40 years in Table 4.1-8 of the USAR. As described above, the numbers of design transient cycles projected to be accumulated at 60 years were less than the numbers of cycles considered in the original 40-year designs. Therefore, the TLAAs for the subject auxiliary heat exchangers will remain valid during the period of extended operation.

Environmental Effects on Fatigue

Generic Safety Issue 190 addressed the issue that certain environmental effects (such as temperature and dissolved oxygen content) in the primary systems of light water reactors could result in greater susceptibility to fatigue than would be predicted by fatigue analyses based on the ASME Section III design fatigue curves. The ASME design fatigue curves were based on laboratory tests in air and at low temperatures. Although the fatigue failure curves derived from laboratory tests were adjusted to account for effects such as data scatter, size effect, and surface finish, these adjustments may not have been sufficient to account for actual plant operating environments.

As reported in SECY-95-245, the NRC concluded that no immediate staff or licensee action was necessary to deal with environmentally-assisted fatigue, and a backfit of the environmental fatigue data to operating plants was not required. However, the NRC also concluded that, because metal fatigue effects increase with service life, environmentally-assisted fatigue should be evaluated for any proposed extended period of operation for License Renewal.

NUREG/CR-6260 applied the fatigue design curves that incorporated environmental effects to several plants and identified locations of interest for consideration of environmental effects. Section 5.5 of NUREG/CR-6260 identified certain component locations to evaluate in older vintage Westinghouse plants, such as PINGP. The corresponding PINGP locations are as follows:

- Reactor vessel shell and lower head
- Reactor vessel inlet and outlet nozzles
- Pressurizer surge line hot leg nozzle safe end
- RCS piping charging system nozzle
- RCS piping safety injection accumulator nozzle
- RHR Class 1 piping tee

For License Renewal the effects of reactor water environment on fatigue were evaluated for the equivalent PINGP locations using the methodology of NUREG/CR-6260. Environmentally-adjusted cumulative usage factors (CUFs) for 60 years were calculated. The environmentally-adjusted CUFs for all locations were projected to be less than 1.0 through the period of extended operation.

A4.3 Environmental Qualification of Electrical Components

The Environmental Qualification of Electrical Components Program manages component thermal, radiation and cyclical aging in accordance with 10 CFR 50.49 through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods. Aging evaluations for Environmentally Qualified components that specify a qualified life of at least 40 years are considered TLAAs for License Renewal.

Aging evaluations of electrical components are updated on an as-required basis to manage the effects of aging on qualified life. When qualification time limits are approached, whether during the initial 40-year license term or the period of extended operation, the Environmental Qualification of Electrical Components Program requires replacement, refurbishment or reanalysis to extend the qualification of components. Therefore, the effects of aging on the intended functions of EQ components will be adequately managed for the period of extended operation.

A4.4 Reactor Containment Vessel and Penetration Fatigue Analyses

The design specification for the Reactor Containment Vessels (RCVs) assumes 40 cycles of pressurization of the vessel from atmospheric pressure to design pressure in 40 years. Because the only time the vessel would experience a pressurization cycle would be for integrated leak rate testing that is typically performed at 10-year intervals, or during certain accident scenarios, the assumption is conservative, and will remain valid through the period of extended operation.

The design specification also assumes 200 temperature cycles between 50°F and 120°F during the life of the vessel. The operating temperature of each RCV stays relatively constant during normal plant operation as the Shield Building effectively isolates the vessel from outdoor weather, and temperature variations are only expected during plant shutdown periods. The temperature variations of the Reactor Containment Vessel can be correlated to plant heat-up and cooldown cycles over 60 years, which are shown in USAR Table 4.1-8 to be limited to 200. Therefore, this assumption will remain valid through the period of extended operation.

Hot piping penetration assemblies, including the process pipe, guard pipe, and flued heads, were designed in accordance with USAS B31.1.0, and can be considered to be subject to the cyclic operation stress range reduction factor. The stress range reduction factor begins to decrease the code allowable stress when the number of thermal cycles become greater than 7,000. The hot piping penetration thermal cycles correlate with Reactor Coolant System heatup and cooldown, and reactor trips. Current USAR allowable cycles for Reactor Coolant System heatup and cooldown and reactor trips are 200 and 400, respectively, which bound the expected number of cycles for the period of extended operation. Therefore, the numbers of applicable design transients will not exceed 7000 cycles in 60 years of plant operation, and this TLAA will remain valid through the period of extended operation.

A4.5 RCS Piping Leak-Before-Break Analyses

Leak-Before-Break (LBB) analyses, discussed in USAR Section 4.6.2.3 and Section 4.6.2.4, evaluate postulated flaw growth in piping to justify changes to the structural design bases involving protection against the effect of postulated reactor coolant pipe ruptures. The LBB evaluations use fully aged fracture toughness properties, and these analyses do not have a material property time-limited assumption. However, the predicted growth of a postulated fatigue crack over 40 years was calculated using the RCS design transients. Since the numbers of design transients accumulated in 60 years remains less than the original 40-year assumptions, these analyses will remain valid during the period of extended operation.

A4.6 Reactor Vessel Underclad Cracking

Intergranular separations (underclad cracking) in low alloy steel heat-affected zones under austenitic stainless steel weld cladding were first detected in SA-508, Class 2, reactor vessel forgings in 1970. They have been reported to exist in SA-508, Class 2, reactor vessel forgings manufactured to a coarse grain practice and clad by high-heat-input submerged arc processes. The subject of underclad cracking is addressed in USAR Section 4.2.3.4.

WCAP-15338 extended the original evaluation of underclad cracking to account for 60 years of operation under a renewed operating license. The numbers of design transient cycles assumed in the WCAP-15338 analysis have been confirmed to bound the numbers of design cycles and transients projected for 60 years of operation at PINGP. Therefore, WCAP-15338 demonstrates for PINGP that fatigue growth of the postulated flaws will be minimal over 60 years, and the presence of underclad cracks are of no concern relative to the structural integrity of the reactor vessels. The analysis of underclad cracking for PINGP remains valid for the period of extended operation.

A4.7 Reactor Coolant Pump Flywheel

As discussed in USAR Section 4.3.3, the reactor coolant pump (RCP) motors are large, vertical, squirrel cage, induction motors. The motors have flywheels to increase rotational-inertia, thus prolonging pump coastdown and retarding the decrease in coolant flow to the core in the event that pump power is lost. The flywheel is mounted on the upper end of the rotor, above the upper radial bearing and inside the motor frame. The aging effect of concern is fatigue crack initiation and growth in the flywheel bore keyway from stresses due to starting the motor.

A license amendment request was submitted in 2004 to reduce the RCP flywheel inspection frequency and scope. The request was based on WCAP-15666, “Extension of Reactor Coolant Pump Motor Flywheel Examination.” This topical report includes a stress and fracture evaluation which adequately addresses fatigue crack growth for 60 years. The NRC approved this request in License Amendments 170 (Unit 1) and 160 (Unit 2) in 2005. Therefore, the analysis of fatigue crack initiation and growth in the RCP flywheels remains valid for the period of extended operation.

A4.8 Fatigue Analysis of Cranes

Design reviews performed in response to NUREG-0612 concluded that the polar cranes, auxiliary building crane, turbine building cranes, and spent fuel crane were qualified to EOCI Specification #61, but are also in compliance with the design standards of CMAA-70, with limited exceptions. Among the criteria of CMAA-70 is a design load cycle limit of 20,000 cycles. (The Class A crane value is limiting.) PINGP has reviewed the usage of these cranes and determined that even very conservative estimates of the number of cycles to be achieved in 60 years of operation do not exceed the 20,000 cycle limit in CMAA-70. As a result, the crane design analyses will remain valid for the period of extended operation.

A4.9 Probability of Damage to Safeguards Equipment from Turbine Missiles

Fully integral turbine rotors were installed at Unit 1 in 1997 and Unit 2 in 1998. A probabilistic evaluation was completed to support the installation that included an evaluation of the probability of damage to safeguards equipment from turbine missiles. A description of the evaluation is contained in Section 12.2.7 of the PINGP USAR.

This probabilistic evaluation is a TLAA for PINGP since periodic in-service inspection frequency of the rotors is based on the requirement that the probability of missile ejection remains below $5x10^{-6}$, and failure probability is a function of time. In accordance with USAR Figure 12.2-38, a failure probability of $5x10^{-6}$ is reached at approximately 37 years after installation, or 2034 for Unit 1 and 2035 for Unit 2. This is beyond the end of the period of extended operation for Unit 1 (2033) and Unit 2 (2034). Therefore, the analysis of probability of turbine rupture due to stress corrosion and the conclusion that periodic inspections are not required remain valid for the period of extended operation.

A5.0 License Renewal Commitments

The preliminary list of commitments made in the Application for Renewed Operating Licenses for Prairie Island Nuclear Generating Plant Units 1 and 2 (LRA) has been provided in the letter transmitting the LRA to the NRC. The commitments reflect the contents of the LRA as submitted, but are considered preliminary in that the specific wording of some commitments may change, and additional commitments may be made, during the NRC review of the LRA. Any other actions discussed in the LRA should be considered intended or planned actions. These other actions are included for informational purposes but are not considered regulatory commitments.

The final commitments as submitted by NMC, and accepted by NRC, are expected to be confirmed in the NRC's Safety Evaluation Report (SER) for the renewed operating licenses. These final commitments, as confirmed in the SER, will become effective upon NRC issuance of the renewed operating licenses. The list of final commitments will be incorporated into the USAR.