CHAPTER 15 TABLE OF CONTENTS

15.0 AGING MANAGEMENT PROGRAMS and TIME LIMITED AGING ANALYSIS

This Section has been selected for the location of Aging Management Program and Time Limited Aging Analysis related information. Section 15.1 provides an overview of the Quality Assurance Program requirements for aging management. Section 15.2 contains a summary description of the programs for managing the effects of aging during the period of extended operation. Time-limited aging analyses (TLAA) supporting activity summaries are contained in Section 15.3. Section 15.4 contains a summary of the evaluation of TLAAs for the period of extended operation. (NUREG-1839, Safety Evaluation Report Related to the License Renewal of the Point Beach Nuclear Plant, Units 1 and 2, Docket Nos. 50-266 and 50-301, December 1, 2005).

15.1 PROGRAMS THAT MANAGE THE EFFECTS OF AGING AND GENERIC QUALITY ASSURANCE PROGRAM REQUIREMENTS

This section provides summaries of the programs and activities credited for managing the effects of aging. These aging management programs may not exist as discrete programs at PBNP. In many cases they exist as a compilation of various implementing documents that, when taken as a whole, satisfy the intent of NUREG-1800 and/or NUREG-1801 elements.

The Quality Assurance Topical Report implements the requirements of 10 CFR 50, Appendix B, and is consistent with the summary in Appendix A.2 of NUREG-1800, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," published July 2001. The elements of corrective action, confirmation process, and administrative controls in the Quality Assurance Program are applicable to both safety-related and nonsafety related systems, structures and components that are subject to an aging management review. Generically, these three elements are applicable as follows:

Corrective Actions

Corrective actions are implemented in accordance with the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," and the Quality Assurance Topical Report.

Confirmation Process

The confirmation process is part of the corrective action program, which is implemented in accordance with the requirements of 10 CFR 50, Appendix B and the Quality Assurance Topical Report.

Administrative Controls

Aging management programs are implemented through various plant documents. These implementing documents are subject to administrative controls, including a formal review and approval process, in accordance with the requirements of 10 CFR 50, Appendix B, and the Quality Assurance Topical Report.

15.2 AGING MANAGEMENT PROGRAM DESCRIPTIONS

The description of the PBNP Aging Management Programs are consistent with their status as configured to apply to the period of extended operation.

15.2.1 ASME SECTION XI, SUBSECTIONS IWB, IWC, AND IWD ISI PROGRAM

ASME Section XI, Subsections IWB, IWC and IWD Inservice Inspection (ISI) Program inspections are performed to identify and correct degradation in Class 1, 2 and 3 piping, components and their integral attachments. The program includes periodic visual, surface, and/or volumetric examinations and leakage tests of Class 1, 2 and 3 pressure-retaining components, and their integral attachments, including welds, pump casings, valve bodies, and pressure-retaining bolting. These components and their integral attachments are identified in ASME Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," or commitments requiring augmented inservice inspections, and are within the scope of license renewal. This program will use the edition and addenda of ASME Section XI required by 10 CFR 50.55a, as reviewed and approved by the NRC staff for aging management under 10 CFR 54. Alternatives to these requirements that are aging management related will be submitted to the NRC in accordance with 10 CFR 50.55a prior to implementation.

15.2.2 ASME SECTION XI, SUBSECTIONS IWE AND IWL ISI PROGRAM

The ASME Section XI, Subsections IWE and IWL Inservice Inspection Program manages aging of (a) steel liners of concrete containments and their integral attachments; containment hatches and airlocks; seals, gaskets and moisture barriers; and pressure-retaining bolting, and (b) reinforced concrete containments and unbonded post-tensioning systems. The primary inspection methods employed are visual examinations with limited supplemental volumetric and surface examinations, as necessary. Tendon anchorages and wires are visually examined. Tendon wires are tested to verify that minimum mechanical property requirements are met. Tendon corrosion protection medium is analyzed for alkalinity, water content and soluble ion concentrations. Pre-stressing forces are measured in sample tendons. Measured tendon lift-off forces are compared to predicted tendon forces calculated in accordance with Regulatory Guide 1.35.1. This program will use the edition and addenda of ASME Section XI required by 10 CFR 50.55a, as reviewed and approved by the NRC staff for aging management under 10 CFR 54. Alternatives to these requirements that are aging management related will be submitted to the NRC in accordance with 10 CFR 50.55a prior to implementation.

This program manages aging effects for:

1. Carbon steel and miscellaneous polymeric materials and components that provide containment pressure boundary/leak-tight barrier function and are tested/inspected in accordance with 10 CFR 50, Appendix J and/or ASME Section XI, Subsection IWE,

2. Containment tendons, and

3. Concrete, which is inspected in accordance with ASME Section XI, Subsection IWL.

15.2.3 ASME SECTION XI, SUBSECTION IWF ISI PROGRAM

The ASME Section XI, Subsection IWF Inservice Inspection Program manages aging effects for Class 1, 2 and 3 component supports. The primary inspection method employed is visual examination. Criteria for acceptance and corrective action are in accordance with ASME Section XI, Subsection IWF. Degradation that potentially compromises the function or load capacity of the support, including bolting, is identified for evaluation. Supports requiring corrective action are re-examined during the next inspection period. This program will use the edition and addenda of ASME Section XI required by 10 CFR 50.55a, as reviewed and approved by the NRC staff for aging management under 10 CFR 54. Alternatives to these requirements that are aging management related will be submitted to the NRC in accordance with 10 CFR 50.55a prior to implementation.

15.2.4 BOLTING INTEGRITY PROGRAM

The Bolting Integrity Program manages the aging effects associated with bolting through the performance of periodic inspections. The program also includes repair/replacement controls for ASME Section XI related bolting and generic guidance regarding material selection, thread lubrication and assembly of bolted joints. The program considers the guidelines delineated in NUREG-1339 for a bolting integrity program, EPRI NP-5769 (Reference 1) (with the exceptions noted in NUREG-1339) for safety related bolting, and EPRI TR-104213 (Reference 2) for non-safety related bolting. The Bolting Integrity Program credits seven separate aging management programs for the inspection of bolting. The seven aging management programs are: (1) ASME Section XI, Subsections IWB, IWC and IWD Inservice Inspection Program, (2) ASME Section XI, Subsections IWE and IWL Inservice Inspection Program, (3) ASME Section XI, Subsection IWF Inservice Inspection Program, (4) Systems Monitoring Program, (5) Structures Monitoring Program, (6) Reactor Vessel Internals Program, and (7) the Periodic Surveillance and Preventive Maintenance Program.

15.2.5 BORAFLEX MONITORING PROGRAM

The Boraflex Monitoring Program has been discontinued since Boraflex is no longer credited in the spent fuel pool criticality analysis. (Reference 4)

15.2.6 BORIC ACID CORROSION PROGRAM

The Boric Acid Corrosion Program manages aging effects for structures and components as a result of borated water leakage. The program requires periodic visual inspection of systems that contain borated water for evidence of leakage or accumulations of dried boric acid. It includes provisions for (a) determination of the principal location or source of the leakage, (b) examination requirements and procedures for locating small leaks, and (c) evaluations and/or corrective actions to ensure that boric acid leakage does not lead to degradation of the leakage source as well as other SSC exposed to the leakage, including mechanical, structural, and electrical items such as bolts, fasteners, piping, cables, cable trays, electrical connectors, etc., which could cause the loss of intended function(s). This program complies with PBNP’s response to NRC GL 88-05.

15.2.7 BURIED SERVICES MONITORING PROGRAM

The Buried Services Monitoring Program manages aging effects on the external surfaces of buried carbon steel, low-alloy steel and cast iron components (e.g., tanks, piping) that are within the scope of license renewal in the Service Water, Fuel Oil, and Fire Protections Systems. This program includes (a) preventive measures to mitigate degradation (e.g., external coatings/wrappings), (b) visual inspections of external surfaces of buried components for evidence of coating/wrapping damage and (c) visual inspections and/or hardness testing of external surfaces of buried components for evidence of degradation, if the coating/wrapping is damaged or the pipe is uncoated/unwrapped, to manage the effects of aging. The periodicity of these inspections will be based on plant operating experience and opportunities for inspection such as scheduled maintenance work. In addition, a susceptible location in the Fire Protection System (i.e., uncoated/unwrapped piping) will be scheduled to be inspected once prior to the period of extended operation and at least every 10 years during the period of extended operation. The intent of these scheduled inspections is to ensure that buried components within the Fire Protection System are periodically inspected. Therefore, if an opportunity for inspection occurs prior to the scheduled inspection, the inspection of opportunity can be credited for satisfying the scheduled inspection.

15.2.8 CABLE CONDITION MONITORING PROGRAM

The Cable Condition Monitoring Program manages aging of conductor insulation materials on cables and connectors, and other electrical insulation materials that are installed in adverse localized environments caused by heat, radiation, or moisture. The scope of this program includes accessible non-EQ electrical cables and connections, including control and instrumentation circuit cables, non-EQ electrical cables used in nuclear instrumentation circuits, and inaccessible non-EQ medium-voltage cables within the scope of license renewal. The program requires (a) visual inspection of a representative sample of accessible electrical cables and connections in adverse localized environments once every 10 years for evidence of jacket surface degradation, (b) testing of nuclear instrumentation circuits once every 10 years to detect a significant reduction in cable insulation resistance, and (c) testing of a representative sample of in-scope, medium-voltage cables not designed for submergence subject to significant moisture and significant voltage once every 10 years to detect deterioration of insulation.

15.2.9 CLOSED-CYCLE COOLING WATER SYSTEM SURVEILLANCE PROGRAM

The Closed-Cycle Cooling Water System Surveillance Program manages aging effects in closed cycle cooling water systems that are not subject to significant sources of contamination, in which water chemistry is controlled and heat is not directly rejected to the ultimate heat sink. The program includes (a) maintenance of system corrosion inhibitor concentrations to minimize degradation, and (b) periodic or one-time surveillance testing and inspections to evaluate system and component performance. Inspection methods may include visual, ultrasonic (UT) and eddy current (ECT) testing.

15.2.10 FIRE PROTECTION PROGRAM

The Fire Protection Program includes (a) fire barrier inspections, (b) electric and diesel-driven fire pump tests, (c) periodic inspection and testing of the halon fire suppression system, and (d) periodic maintenance, testing, and inspection of water-based fire protection systems. Periodic visual inspections of fire barrier penetration seals, fire dampers, fire barrier walls, ceilings and floors, and periodic visual inspections and functional tests of fire-rated doors are performed to ensure that functionality and operability is maintained. Periodic testing of the electric and diesel-driven fire pumps ensures that an adequate flow of firewater is supplied and that there is no degradation of diesel fuel supply lines. Periodic maintenance, testing and inspection activities of water-based fire protection systems provides reasonable assurance that fire water systems are capable of performing their intended function. Inspection and testing is performed in accordance with the nuclear insurance carrier's fire protection system testing requirements and generally follows the guidance of applicable NFPA Codes and Standards, as described in the PBNP Fire Protection Evaluation Report.

15.2.11 FLOW-ACCELERATED CORROSION PROGRAM

The Flow Accelerated Corrosion Program manages aging effects due to flow-accelerated corrosion on the internal surfaces of carbon or low alloy steel piping, elbows, reducers, expanders, and valve bodies which contain high energy fluids (both single phase and two phase). The program implements the EPRI guidelines in NSAC-202L-R3 (Reference 3) for an effective FAC program and includes (a) an analysis using a predictive code such as CHECWORKS to determine critical locations, (b) baseline inspections to determine the extent of thinning at these locations, (c) follow-up inspections to confirm the predictions, and (d) repairing or replacing components, as necessary.

15.2.12 FUEL OIL CHEMISTRY CONTROL PROGRAM

The Fuel Oil Chemistry Control Program mitigates and manages aging effects on the internal surfaces of fuel oil storage tanks and associated components in systems that contain fuel oil. The program includes (a) surveillance and monitoring procedures for maintaining fuel oil quality by controlling contaminants in accordance with applicable ASTM Standards, (b) periodic draining of water from fuel oil tanks, (c) periodic or conditional visual inspection of internal surfaces or wall thickness measurements (e.g., by UT) from external surfaces of fuel oil tanks, and (d) one-time inspections of a representative sample of components in systems that contain fuel oil.

15.2.13 ONE-TIME INSPECTION PROGRAM

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

The program elements include (a) determination of appropriate inspection sample size, (b) identification of inspection locations, (c) selection of examination technique, with acceptance criteria, and (d) evaluation of results to determine the need for additional inspections or other corrective actions. The inspection sample includes locations where the most severe aging effect(s) would be expected to occur. Inspection methods may include visual (or remote visual), surface or volumetric examinations, or other established NDE techniques.

This program is used for the following:

- To verify the effectiveness of water chemistry control for managing the effects of aging in stagnant or low-flow portions of piping, or occluded areas of components, exposed to a treated water environment.
- To manage the aging effects of loss of material due to galvanic corrosion and selective leaching.
- To verify that in areas not managed by a chemistry control program the aging effects are occurring so slowly that the intended function will be unaffected through the period of extended operation.
- To verify the effectiveness of fuel oil chemistry control for managing the effects of aging of various components in systems that contain fuel oil.
- To verify aging effects are not occurring in various components (e.g., reactor vessel internals hold-down spring, letdown orifices, steam traps, downstream piping near the RHR pumps' mini-flow recirculation orifices, and miscellaneous heat exchangers).

15.2.14 OPEN-CYCLE COOLING (SERVICE) WATER SYSTEM SURVEILLANCE PROGRAM

The Open-Cycle Cooling (Service) Water System Surveillance Program manages aging effects caused by exposure of internal surfaces of metallic components in water systems (e.g., piping, valves, heat exchangers) to raw, untreated (e.g., service) water. The aging effects are managed through (a) surveillance and control of biofouling, (b) verification of heat transfer by testing, and (c) routine inspection and maintenance program activities to ensure that aging effects do not impair component intended function. Inspection methods include visual, ultrasonic (UT), eddy current (ECT), and Tangential Radiography. This program complies with the licensee’s response to NRC Generic Letter 89-13 and subsequent commitment changes.

15.2.15 PERIODIC SURVEILLANCE AND PREVENTIVE MAINTENANCE PROGRAM

The Periodic Surveillance and Preventive Maintenance Program manages aging effects for certain SSCs within the scope of license renewal. The program provides for inspection, examination, or testing of selected structures and components, including fasteners, for evidence of age-related degradation on a specified frequency based on operating experience or other requirements (e.g., Technical Specification or Code requirements). Additionally, the program provides for replacement of certain components on a specified frequency based on operating experience. The Periodic Surveillance and Preventive Maintenance Program is also used to verify the effectiveness of other aging management programs.

15.2.16 REACTOR COOLANT SYSTEM ALLOY 600 INSPECTION PROGRAM

The Reactor Coolant System Alloy 600 Inspection Program manages crack initiation and growth due to PWSCC of RCS pressure boundary and non-pressure boundary nickel-based alloy components (e.g., Alloy 600/690 reactor vessel/head penetration nozzles, Inconel 82/182, 82/152, and 52/152 weld joints). The program includes (a) PWSCC susceptibility assessment using industry models to identify susceptible components, (b) monitoring and control of reactor coolant chemistry to mitigate PWSCC, (c) inservice inspections (ISI) of reactor vessel/head penetrations and RCS pressure boundary welds in accordance with ASME Section XI, Subsection IWB, Table IWB 2500-1, and (d) augmented inspections or preemptive repair/replacement of susceptible components or welds. The program is based on the guidance provided in Electric Power Research Institute (EPRI) MRP-126 “Generic Guidance for Alloy 600 Management.” (Reference 5)

15.2.17 REACTOR VESSEL INTERNALS PROGRAM

The Reactor Vessel Internals Program manages the aging effects for reactor vessel internals (RVI). The program provides for (a) Inservice Inspection (ISI) in accordance with ASME Section XI requirements, including examinations performed during the 10-year ISI examination; (b) An evaluation that will identify leading locations with respect to IASCC and irradiation embrittlement, appropriate non-destructive examination techniques, and an examination schedule for these locations; (c) Baffle-former/barrel-former bolt evaluation that will determine the acceptability of the current arrangement or if ultrasonic examination and/or replacement of these bolts is necessary; (d) For cast austenitic stainless steel components subject to neutron fluence in excess of 1E17 n/cm^2 or determined to be susceptible to thermal embrittlement, an augmented inspection of components experiencing significant tensile stress (>5 ksi); (e) Evaluation of the significance of void swelling; (f) monitoring and control of reactor coolant water chemistry in accordance with the Water Chemistry Control Program to mitigate SCC or IASCC; (g) Participation in industry initiatives that will generate additional data on aging mechanisms relevant to RVI and develop appropriate inspection techniques to permit detection and characterization of features of interest; and (h) One-time inspection of the internals hold-down spring for evidence of stress relaxation. (Reference 6)

15.2.18 REACTOR VESSEL SURVEILLANCE PROGRAM

The Reactor Vessel Surveillance Program manages the aging effect reduction of fracture toughness due to neutron embrittlement of the low alloy steel reactor vessels. Monitoring methods will be in accordance with 10 CFR 50, Appendix H. This program includes (a) capsule insertion, withdrawal and materials testing/evaluation, (including upper shelf energy and RTNDT determinations), (b) fluence and uncertainty calculations, (c) monitoring of Effective Full Power Years (EFPY), (d) development of pressure-temperature limitations, (e) determination of low-temperature overpressure protection (LTOP) set points, and (f) implementation of a flux reduction program and other options, as necessary, allowed by 10 CFR 50.61(b) for the Unit 2 intermediate-to-lower shell girth weld. The program ensures the reactor vessel materials (a) meet the fracture toughness requirements of 10 CFR 50, Appendix G, and (b) have adequate margins against brittle fracture caused by Pressurized Thermal Shock (PTS) in accordance with 10 CFR 50.61.

15.2.19 STEAM GENERATOR INTEGRITY PROGRAM

The Steam Generator Integrity Program incorporates the guidance of NEI 97-06 and maintains the integrity of the steam generators, including tubes, tube plugs or other tube repairs, and various secondary-side internal components. The program manages aging effects through a balance of prevention, inspection, evaluation, repair, and leakage monitoring measures. Component degradation is mitigated by controlling primary and secondary water chemistry. Eddy current testing is used to detect steam generator tube flaws and degradation. Visual inspections are performed to identify degradation of various secondary-side steam generator internal components.

15.2.20 STRUCTURES MONITORING PROGRAM

The Structures Monitoring Program manages the aging effects associated with steel (including fasteners), concrete (including masonry block and grout), earthen berms, and elastomers. The environments include below grade and fluid exposed material, outdoor weather, and indoor air. The program includes all safety related buildings, structures within the containment, other buildings within the scope of license renewal, crane bridge and trolley structures, and component supports (including HELB structures, panels, etc.) within the scope of license renewal. The program provides for periodic visual inspections and examination of accessible surfaces of the structures and components and identifies the aging effects that impact the materials of construction. The program also visually examines normally inaccessible below grade concrete when it is exposed by excavation (i.e., inspections of opportunity) for signs of degradation.

15.2.21 SYSTEMS MONITORING PROGRAM

The Systems Monitoring Program manages aging effects on the external surfaces of piping, tanks and other components and equipment within the scope of license renewal. These aging effects are managed through visual inspection and monitoring of normally accessible external surfaces for leakage and evidence of material degradation.

15.2.22 TANK INTERNAL INSPECTION PROGRAM

The Tank Internal Inspection Program manages aging effects on the (a) internal surfaces of carbon steel tanks, and (b) inaccessible external surfaces of carbon steel tanks (i.e., tank bottoms) where wall thickness measurements may be taken from inside the tank to detect external degradation (e.g., using ultrasonic techniques)

This program provides for periodic inspections to confirm that aging effects will not impair tank intended functions. Tank wall thinning of internal surfaces may be detected by direct visual inspection from inside the tank or indirectly by UT wall thickness measurements from outside the tank. Tank wall thinning of external surfaces that are inaccessible (e.g., bottom of tanks that sit directly on the ground or other support structures) will be detected by UT wall thickness measurements from inside the tank.

15.2.23 THIMBLE TUBE INSPECTION PROGRAM

The Thimble Tube Inspection Program manages aging effects for incore instrument thimble tubes. This program requires periodic eddy current testing of thimble tubes and contains criteria for determining sample size, inspection frequency, flaw evaluation, and corrective action, in accordance with NRC Bulletin 88-09.

15.2.24 WATER CHEMISTRY CONTROL PROGRAM

The Water Chemistry Control Program manages aging effects by controlling the internal environment of systems and components. Primary borated and secondary water systems are included in the scope of the program. The program is based on EPRI PWR Primary Water Chemistry Guidelines and EPRI PWR Secondary Water Chemistry Guidelines. Guideline revisions are evaluated for applicability to PBNP and program implementing documents are revised as necessary. The aging effects are managed by controlling concentrations of known detrimental chemical species such as halogens, 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. For low-flow or stagnant portions of a system, a one-time inspection of selected components at susceptible locations provides verification of the effectiveness of the Water Chemistry Control Program. No verification inspections are required for intermediate and high flow regions.

15.2.25 REFERENCES

1. EPRI NP-5769, "Degradation and Failure of Bolting in Nuclear Power Plants," dated April 1988. (Volume I and Volume II)

2. EPRI TR-104213, "Bolted Joint Maintenance & Applications Guide," dated December 1995.

3. EPRI Nuclear Safety Analysis Center NSAC 202L, "Recommendations for an Effective Flow Accelerated Corrosion Program," Revision 3, dated August 2007.

4. NRC Safety Evaluation, "Point Beach Nuclear Plant, Units 1 and 2 - Issuance of Amendments Re: Spent Fuel Pool Storage Criticality Control," dated March 5, 2010.

5. NRC Safety Evaluation, Point Beach Nuclear Plant, Units 1 and 2 -Alloy 600 Program License Renewal Commitment Submittal, dated October 6, 2009.

6. Point Beach Nuclear Plant, Units 1 and 2, Staff Assessment of Reactor Vessel Internals Inspection Plan Based on MRP-227-A (TAC NOS. ME8235 and ME8236), dated March 30, 2015.

15.3 TIME LIMITED AGING ANALYSIS SUPPORTING ACTIVITIES

Environmental Qualification Program

The EQ Program manages component thermal, radiation and cyclical aging, as applicable, through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods. As required by 10 CFR 50.49, EQ components not qualified for the current license term are to be refurbished, replaced, or have their qualification extended prior to reaching the aging limits established in the evaluation. Aging evaluations for EQ components that specify a qualification of at least 40 years are considered TLAA for license renewal. The EQ Program ensures that these EQ components are maintained within the bounds of their qualification bases.

Fatigue Monitoring Program

The Fatigue Monitoring Program is a confirmatory program that monitors loading cycles due to thermal and pressure transients and cumulative fatigue usage for selected reactor coolant system and other component locations. The program provides an analytical basis for confirming that the actual number of cycles does not exceed the number of cycles used in the design analysis, and the cumulative fatigue usage will be maintained below the allowable limit during the period of extended operation.

The impact of the effects of reactor coolant environment on component fatigue life has been evaluated for a sample of critical components, including the seven component locations selected in NUREG/CR-6260. Appropriate environmental fatigue factors were calculated using the formulae from NUREG/CR-6583 for carbon and low-alloy steels and NUREG/CR-5704 for austenitic stainless steels. These critical component locations were determined to be acceptable for the period of extended operation, including the effects of reactor coolant environment. The acceptability of these critical component locations, including the effects of reactor coolant environment, will continue to be confirmed by the Fatigue Monitoring Program.

15.4 EVALUATION OF TIME-LIMITED AGING ANALYSES

As part of a License Renewal Application, 10 CFR 54.21(c) requires that an evaluation of time-limited aging analyses (TLAAs) for the period of extended operation be provided. The following TLAAs have been identified and evaluated to meet this requirement. These discussions are numbered and inserted into the FSAR sections where these subjects are covered.

During the Extended Power Uprate Project the TLAAs were evaluated for operation at 1800 megawatts thermal (Reference 17). References to evaluations at lower power levels completed as part of the License Renewal Program were retained for their historical perspective.

15.4.1 REACTOR VESSEL IRRADIATION EMBRITTLEMENT

The PBNP Units 1 and 2 reactor vessels are described in Chapter 3.0 and Chapter 4.0. Time-limited aging analyses (TLAAs) applicable to the reactor vessels are:

- Pressurized thermal shock
- Upper-shelf energy
- Pressure-temperature limits

The Reactor Vessel Surveillance Program manages reactor vessel irradiation embrittlement utilizing subprograms to monitor, calculate, and evaluate the time-dependent parameters used in the aging analyses for pressurized thermal shock, upper-shelf energy and pressure-temperature limit curves to ensure continuing vessel integrity through the period of extended operation.

Reactor Vessel Pressurized Thermal Shock

The requirements in 10 CFR 50.61 provide rules for protection against pressurized thermal shock events for pressurized water reactors. Licensees are required to perform an assessment of the projected values of the maximum nil ductility reference temperature (RT_{PTS}) whenever a significant change occurs in projected values of RT_{PTS}, or upon request for a change in the expiration date for the operation of the facility.

The calculated RT_{PTS} values at the end of life extension for the PBNP Units 1 and 2 reactor vessels are less than the 10 CFR 50.61(b)(2) screening criteria of 270°F for intermediate and lower shells and 300°F for the circumferential welds. Initially, the Unit 2 RPV intermediate to lower shell circumferential weld exceeded the screening criteria established in 10 CFR 50.61 (300°F) during the period of extended operation.

Amendment Nos. 250 and 254 (Reference 16) allow the use of an alternate fracture toughness evaluation methodology for determining RCS pressure and temperature limits. BAW-2308, Revision 1-A and 2-A, “Initial RT_{NDT} of Linde 80 Weld Materials,” provides an alternative estimation of the initial nil-ductility reference temperature (RT_{NDT}) of Linde 80 weld materials. With this “Master Curve” methodology, the Unit 2 intermediate-to-lower shell circumferential weld does not exceed the PTS screening criteria at End-of-Life-Extended (EOLE).

Reactor Vessel Upper-Shelf Energy

The requirements on reactor vessel Charpy upper-shelf energy (USE) are included in 10 CFR 50, Appendix G. Specifically, 10 CFR 50, Appendix G requires licensees to submit an analysis at least 3 years prior to the time that the upper-shelf energy of any reactor vessel material is predicted to drop below 50 ft-lb, as measured by Charpy V-notch specimen testing.

The Charpy USE for the limiting welds will be less than 50 ft-lbs based on RG 1.99, Revision 2, at 53 effective full power years (EFPY). Therefore, in order to demonstrate that sufficient margins of safety against fracture remain to satisfy the requirements of Appendix G to 10 CFR Part 50, a fracture mechanics evaluation was performed to examine the PBNP USE values in the limiting weld. The evaluation examined the USE values for end of license extension (EOLE) conditions. The PBNP fracture mechanics evaluation used the J-R ratio methodology, which demonstrates the acceptability of J-R values in satisfying the USE requirement by examining J-R ratios, which are defined as the ratio of the lower bound J-R value divided by the applied J. If this ratio is greater than or equal to one, the acceptance criteria are met. This methodology is described in B&W Owners Reactor Vessel Working Group reports BAW-2192PA, “Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Level A & B Service Loads,” and BAW-2178-PA, “Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Level C & D Service Loads,” both dated April 1994. The NRC staff reviewed and approved both reports for referencing in licensing applications in separate letters dated March 29, 1994.

Additional equivalent margins analyses were performed for the PBNP RPVs to address the following EOLE (53 EFPY) conditions: the uprated power condition of 1678 megawatts thermal (MWt) without hafnium suppression assemblies; current power conditions of 1540 MWt without hafnium suppression assemblies; and current power conditions of 1540 MWt with hafnium suppression assemblies. The 2008 fluence projections (Reference 9) were used to define EOLE vessel fluences at 1800 MWt. An equivalent margins analysis compared the EPU fluence values with the fluence values used in BAW-2467P, Revision 1 (Reference 10). This analysis demonstrated that the analysis in BAW-2467P, Revision 1, remains applicable for the projected EPU fluence values to the end of the period of extended operation (Reference 20). The NRC reviewed and accepted the BAW-2467P, Revision 1, analysis in the Safety Evaluation Report transmitted by NRC letter dated May 10, 2007 (Reference 11).

The commitment related to the PPSAs for Unit 2 from NUREG-1839 (Reference 12, Appendix A, Commitment #46) “Safety Evaluation Report Related to the License Renewal of the Point Beach Nuclear Plants Units 1 and 2,” no longer applies due to Amendment Nos. 250 and 254 (Reference 16). These amendments allow the use of BAW-2308, Revision 1-A and 2-A, “Initial RT_{NDT} of Linde 80 Weld Materials” as an alternate fracture toughness evaluation methodology for determining RCS pressure and temperature limits. With the “Master Curve” methodology, the calculated RT_{PTS} values at the end of life extension for the Unit 2 reactor vessel intermediate-to-lower shell circumferential weld is less than the 10 CFR 50.61(b)(2) screening criteria of 300°F, without crediting PPSAs.

The analysis associated with upper-shelf energy has been projected to the end of the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

Reactor Vessel Pressure/Temperature Limits

The requirements in 10 CFR 50, Appendix G, ensure that heatup and cooldown of the reactor pressure vessel are accomplished within established pressure-temperature limits. These limits specify the maximum allowable pressure as a function of reactor coolant temperature. As the reactor pressure vessel becomes embrittled and its fracture toughness is reduced, the allowable pressure is reduced.

Operation of the Reactor Coolant System is also limited by the net positive suction head curves for the reactor coolant pumps. These curves specify the minimum pressure required to operate the reactor coolant pumps. Therefore, in order to heatup and cooldown, the reactor coolant temperature and pressure must be maintained within an operating window established between the Appendix G pressure-temperature limits and the reactor coolant pumps net positive suction head curves.

To address the period of extended operation, the end of license extension projected fluences, and the RPV material properties were used to determine the limiting materials, and calculate pressure-temperature limits for heatup and cooldown. The Point Beach Unit 1 and 2 heatup and cooldown pressure-temperature limit curves were generated using adjusted reference temperature (ART) values that bound both units (Reference 21). The term of applicability for the pressure-temperature curves is 50 EFPY under uprated conditions (1800 MWt) (Reference 16).

The analysis associated with reactor vessel pressure-temperature limit curves has been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

15.4.2 FATIGUE

The thermal and mechanical fatigue analyses of plant mechanical components have been identified as time limited aging analyses for the Point Beach Nuclear Plant. Specific components have been designed and analyzed considering transient cycle assumptions identified in vendor specifications and the PBNP FSAR.

In conjunction with revising the NSSS design transients for the Unit 2 Replacement Steam Generator Project (SGRP), and the Extended Power Uprate Project, the NSSS design transients were also evaluated for acceptability for a 60-year operating period. The number of NSSS transients actually experienced by the two units was identified. Based on historical transient occurrences, and current plant operational practices, the number of future NSSS transients was forecasted for a 60-year operating period. With few exceptions, the anticipated number of transients for a 60-year operating period was far less than the original design number of transients for a 40-year operating period.

The exceptions noted above comprise a set of pressure test transients that were included in some of the NSSS component equipment specifications. The pressure test transients forecasted for a 60-year operating period exceeded the original design number of transients for a 40-year operating period. The NSSS design transient set was revised to include an increased number of pressure test transients, sufficient for a 60-year operating period.

In addition, the NSSS transient set was also revised to increase the number of steady-state random RCS pressure and temperature fluctuations to ensure adequate margin existed for a 60-year operating period. The revised set of NSSS design transients were used in performing the detailed engineering evaluations in support of the Extended Power Uprate Project (Reference 17).

Experience has shown, however, that actual plant operation is often very conservative with respect to the design transients. The use of actual operating history and transient monitoring data acquired by the FatiguePro Automatic Cycle Counting and Fatigue Monitoring System installed at Point Beach (Fatigue Monitoring Program) will allow quantification of the conservatism in the existing fatigue analysis and demonstrate that the design fatigue analyses will bound the extended period of operation. The PBNP Fatigue Monitoring Program is considered a confirmatory program.

ASME Boiler and Pressure Vessel Code, Section III, Class 1 Components

The PBNP Reactor Vessels, CRDMs, Steam Generators, and Pressurizers were designed, constructed and analyzed to the requirements of their original equipment specifications, and Section III of the ASME Code. The PBNP Reactor Vessels Internals and Reactor Coolant Pumps were designed, constructed and analyzed to the requirements of their original equipment specifications, and the intent of Section III of the ASME Code.
The fatigue calculations were reanalyzed for the above noted components at Extended Power Uprate conditions using the revised transient set for a 60-year operating period. The structural evaluations concluded that all components analyzed for fatigue are within the allowable limits for a 60-year operating period, with the exception of the Unit 1 Steam Generator inspection port bolts. The structural evaluation identifies a replacement interval of 12 years for the inspection port bolts.

In addition to the original ASME CLB analysis, a plant-specific insurge/outsurge fatigue analysis was performed for the extended license period. The analysis demonstrated acceptable structural integrity for the affected pressurizer locations to the end of license extension.

With the exception of the Unit 1 steam generator inspection port bolting, the analyses associated with verifying the structural integrity of the PBNP ASME III Class 1 components have been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

The Periodic Surveillance and Preventive Maintenance Program will provide reasonable assurance that the Unit 1 SG inspection port bolt replacement is adequately managed for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(iii).

Pressurizer Surge Line Structural Integrity

Detailed fatigue analyses of the pressurizer surge lines were performed in response to NRC Bulletin 88-11, “Pressurizer Surge Line Thermal Stratification.” The analyses were performed in accordance with the requirements of Section III of the ASME Code. The methodology and results are presented in WCAP-13509, “Structural Evaluation of the Point Beach Units 1 & 2 Pressurizer Surge Lines, Considering the Effects of Thermal Stratification.”

Subsequently, the PBNP-specific surge line fatigue analysis was re-evaluated considering the operational conditions associated with the Extended Power Uprate and a 60-year operating period. The transient sets were reviewed for the new conditions. The majority of the transients defined for original power levels for 40 years were found to be bounding for EPU conditions for 60 years. Some of the feedwater transients required minor revision due to a change in feedwater temperatures associated with the proposed power uprate. The impact of the changes in the revised RCS conditions, thermal design transients, and the 60-year life were factored into determining the ASME stress levels and allowables for the surge line.

The results of the evaluation for the pressurizer surge line stratification showed that the EPU conditions changed the fatigue usage factors at the location of the highest usage factor by a negligible amount. The calculated change in the loadings on the pressurizer nozzle due to stratification for the EPU conditions was not considered significant. The results of the original evaluation for the surge line, WCAP-13509, remain unchanged for the 60-year operating period.

The analysis associated with verifying the structural integrity of the pressurizer surge line piping has been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

Spray Header Piping Structural Integrity

Piping connections to the RCS were evaluated in response to NRC Bulletin 88-08 (including Supplements 1 through 3) “Thermal Stresses in Piping Connected to Reactor Coolant Systems.” Two unisolable piping connections were identified that had the potential to be subjected to thermal stratification or temperature oscillations. These lines are the auxiliary charging connection, and auxiliary spray connection. These lines were subject to temperature monitoring to identify and quantify thermal stratification. No thermal stratification was noted on the auxiliary charging lines. Thermal stratification was noted on one of the auxiliary spray lines, where it ties into the spray header.

To evaluate the effect of thermal stratification on the pressurizer spray line header, including the auxiliary spray line connection, fatigue analyses were performed for each unit's applicable piping system. The analyses were based on actual piping surface temperature data obtained during a 153-day period (including one startup) of direct temperature monitoring on the Unit 2 piping. The Unit 2 data was considered applicable and bounding for both units since it experienced more stratification, and the line configuration was similar. The piping transient set was developed by expanding the measured piping thermal behavior to equate to a 60-year operating period. The analyses showed that the Cumulative Usage Factors (CUFs) in the subject piping were acceptable.

The analysis associated with verifying the structural integrity of the pressurizer auxiliary spray line, and spray header, have been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii). An additional evaluation determined that all ASME Code stress limits remain satisfied for all proposed EPU conditions (Reference 20).

USAS B31.1 Piping Structural Integrity

In general, piping and associated pressure boundary components at PBNP were originally designed to the requirements of USAS B31.1, USA Standard Code for Pressure Piping. The B31.1

Code requirements assume a stress range reduction factor to provide conservatism in the piping design to account for the effects of thermal fatigue due to thermal cycling during operation. This reduction factor is 1.0 provided that the number of anticipated cycles is limited to 7000 equivalent full temperature cycles. This represents a condition where a piping system would have to be cycled approximately once every 3 days over the extended plant life of 60 years. Considering this limit, a review of the piping and associated pressure boundary components was performed to identify those systems that operate at elevated temperature and to establish their cyclic operating practices. Under current plant operating practices, piping is generally only occasionally subject to cyclic operation. Typically, piping is subject to continuous steady state operation and operating temperatures only vary during plant heatup and cooldown, during plant transients or during periodic testing. It is therefore very unlikely, for any piping system subject to thermal fatigue, that the actual number of thermal cycles would approach the assumed B31.1 limit of 7000 during the period of extended operation except for the Primary Sampling System lines. Establishing sample flow from the RCS results in thermal transients and cyclic stresses whenever the RCS is above ambient temperatures. The hot leg sample line receives the highest number of thermal cycles of all PBNP piping. An evaluation of the number of thermal cycles that the hot leg sample line would be expected to experience over a 60 year period of operation was performed in PBNP License Renewal Technical Report, LR-TR 516. The Technical Report demonstrates that the PBNP hot leg sample line will not exceed 7000 thermal cycles over a 60 year operating period. Thus, no PBNP piping is expected to exceed 7000 thermal cycles over a 60 year operating period, and thus remain within the bounds of their original design code.

The analyses associated with USAS B31.1 piping fatigue have been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i). The USAS B31.1 Code does not require a fatigue evaluation of the Reactor Coolant Loop piping system for proposed EPU conditions (Reference 17 and Reference 20).

Environmental Effects on Fatigue

As a part of the industry effort to address environmental effects for operating nuclear power plants during the current 40-year licensing term, Idaho National Engineering Laboratories (INEL) evaluated, in NUREG/CR-6260 (Reference 4), fatigue-sensitive component locations at plants designed by all four U. S. nuclear steam supply system vendors. The pressurized water reactor calculations, especially the early-vintage Westinghouse calculations, are directly relevant to PBNP. In addition, the transient cycles considered in the evaluation match or bound the PBNP design.

The fatigue-sensitive component locations chosen in NUREG/CR-6260 for the early-vintage Westinghouse plant were:

1. The reactor vessel shell and lower head
2. The reactor vessel inlet and outlet nozzles
3. The pressurizer surge line (including the pressurizer and hot leg nozzles)
4. The Reactor Coolant System piping charging system nozzle
5. The Reactor Coolant System piping safety injection nozzle
6. The Residual Heat Removal System Class 1 piping.

In addition to the NUREG/CR-6260 locations, the PBNP pressurizers were evaluated for the effects of coolant environment on fatigue, including insurge/outsurge transients, in accordance with Applicant Action Item 3.3.1.1 1 of the pressurizer Generic Technical Report WCAP-14574-A.

Environmental fatigue evaluations were performed for the NUREG/CR-6260 component locations, and the pressurizers using the Fen methodology contained in NUREG/CR-6583 for carbon/low alloy steel material and NUREG/CR-5704 for stainless steel material.

The effects of reactor coolant environment on component fatigue life during the period of extended operation have been evaluated at PBNP. The evaluation includes the seven component locations identified in NUREG/CR-6260, and the Pressurizer. Appropriate environmental fatigue factors have been applied to either the components design cumulative fatigue usage factor, or the components forecasted cumulative fatigue usage factor, based on actual operational transient monitoring by the EPRI FatiguePro software. The evaluations result in acceptable environmentally adjusted cumulative fatigue usage factors at EOLE for all of the component locations considered. Environmental effects on fatigue during EPU conditions were evaluated and found to be bounded by existing evaluations (Reference 17).

The Fatigue Monitoring Program is a confirmatory program that monitors loading cycles due to thermal and pressure transients for selected critical components. The program provides an analytical basis for confirming that the number of cycles established by the analysis of record will not be exceeded before the end of the period of extended operation.

Containment Liner Plate and Penetrations Fatigue Analysis

The interior surface of each Containment is lined with welded steel plate to provide an essentially leak tight barrier. Design criteria are applied to the liner to assure that the specified allowed leak rate is not exceeded under the design basis accident conditions. The fatigue loads as described in FSAR Section 5.1, were considered in the design of the liner plates and are considered time limited aging analyses for the purposes of license renewal. Each of these has been evaluated for the period of extended operation.

The number of thermal cycles due to annual outdoor temperature variations was increased from 40 to 60 for the extended period of operation. The effect of this increase is insignificant in comparison to the assumed 500 thermal cycles due to Containment interior temperature varying during heatup and cooldown of the Reactor Coolant System. The 500 thermal cycles includes a margin of 300 thermal cycles above the 200 Reactor Coolant System allowable design heatup and cooldown cycles, which is sufficient margin to accommodate the additional 20 cycles of annual outdoor temperature variation. Therefore, this loading condition is considered valid for the period of extended operation as it is enveloped by the evaluation for 500 thermal cycles.

The assumed 500 thermal cycles was evaluated based on the more limiting heatup and cooldown design cycles (transients) for the Reactor Coolant System. The Reactor Coolant

System was designed to withstand 200 heatup and cooldown thermal cycles. The evaluation determined that the originally projected number of maximum Reactor Coolant System design cycles is conservative enough to envelop the projected cycles for the extended period of operation. Therefore, the original containment liner plate fatigue analysis for 500 heatup and cooldown cycles is considered valid for the period of extended operation.

The assumed value of one for thermal cycling due to the maximum hypothetical accident remains valid. No maximum hypothetical accident has occurred and none is expected, therefore, this assumption is considered valid for the period of extended operation.

The design of the containment penetrations has been reviewed. The design meets the general requirements of the 1965 Edition of ASME Boiler and Pressure Vessel Code, Section III. The main steam piping, feedwater piping, blowdown piping, and letdown piping are the only piping penetrating the containment wall and liner plate that contribute significant thermal loading on the liner plate. The projected number of actual operating cycles for these piping systems through 60 years of operation was determined to be less than the original design limits.

The analyses associated with the containment liner plate and penetrations have been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

Crane Load Cycle Limit

The Containment Polar Cranes, Auxiliary Building Crane, and Turbine Hall Crane are included within the scope of license renewal and NUREG-0612.

The load cycle limit for PBNP cranes was identified as a time-limiting-aging analysis.

All PBNP cranes were designed and constructed to meet the requirements of Specification 61 of the Electric Overhead Crane Institute (EOCI-61).

NUREG-0612 required that the design of heavy load overhead handling systems meet the intent of Crane Manufactures Association of America, Inc. (CMAA) Specification No. 70. Per Guideline 7, NUREG-0612, Section 5.1.1(7), the design of the PBNP cranes listed above was evaluated in relation to the applicable criteria and guidelines of Chapter 2-1 of ANSI B30.2-1976, “Overhead and Gantry Cranes,” and of CMAA-70, “Specifications for Electric Overhead Traveling Cranes.” The PBNP cranes listed above substantially meet the criteria of CMAA-70 “Specifications for Electric Overhead Traveling Cranes,” as noted in the NRC NUREG-0612 safety evaluation. Cranes designed in accordance with CMAA-70 Class “A” service are designed for 20,000 to 200,000 load cycles.

The PBNP containment polar cranes and the turbine hall crane are used primarily during refueling outages. The PBNP auxiliary building crane is primarily used in support of material receipt (fuel and consumables), spent fuel cask transfers, and radwaste cask transfers. Occasionally, these cranes make lifts at or near their rated capacity. However, the majority of the crane lifts are substantially less than their rated capacity. Based on conservative usage assumptions, the above listed PBNP cranes are expected to make 50,000 partial load lifts over a 60-year operating period. This is significantly less than the CMAA-70 design cycle limit for Class "A" service cranes.

The specifications for the noted traveling cranes at PBNP included rated overload cycle limits of roughly two 125 percent rated load lifts per year, and three 150 percent rated load lifts in the cranes' lifetimes. With the exception of the containment polar cranes, no lifts in excess of the rated load have been made. Each containment polar crane was used to support its respective units steam generator replacement project. These lifts incorporating the containment polar cranes were specifically analyzed engineered lifts incorporating temporary replacement trolleys, bridge strengthening, and temporary center poles to ensure that the original design capabilities of the cranes were not degraded. Thus, since the major cranes are not used to make routine over rated load lifts, and special one-time over rated load maintenance lifts are addressed as specific engineered lifts, the original specified cycle limits for over rated load lifts will not be exceeded during the extended operating period.

Since the number of operating load cycles for the cranes will be fewer than the design cycles, the crane design will remain valid for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

15.4.3 FRACTURE MECHANICS ANALYSIS

Reactor Coolant Pump Flywheel Analysis

During normal operation, the reactor coolant pump flywheel possesses sufficient kinetic energy to potentially produce high-energy missiles in the unlikely event of failure. Conditions that may result in overspeed of the reactor coolant pump increase both the potential for failure and the kinetic energy. The aging effect of concern is fatigue crack initiation in the flywheel bore keyway.

An evaluation of the probability of failure over the extended period of operation was performed in WCAP-14535-A, Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination, for all operating Westinghouse plants and certain Babcock and Wilcox plants. It demonstrates that the flywheel design has a high structural reliability with a very high flaw tolerance and negligible flaw crack extension over a 60-year service life. The NRC reviewed and approved the evaluation (WCAP-14535-A) for application with certain conditions and limitations (Reference 1). PBNP verified the RCP flywheel material and invoked this analysis as the basis for reducing the frequency of performing RCP flywheel inspections (Reference 2).

WCAP-15666-A, Revision 1, "Extension of Reactor Coolant Pump Motor Flywheel Examination," October 2003, builds on the arguments in WCAP-14535-A and provides additional rationale, including a risk assessment of all credible flywheel speeds. WCAP-15666-A concludes that the change in risk is below Regulatory Guide 1.174 CDF and LERF acceptable guidelines.

The NRC approved the use of this Topical Report in NRC SER, "Safety Evaluation of Topical Report WCAP-15666, Extension of Reactor Coolant Pump Motor Flywheel Examination," May 5, 2003. The NRC SER has been incorporated into the "A" revision of the WCAP. This analysis was used as a basis for a revision of TS 5.5.6 which increased the flywheel inspection interval from 10 years to 20 years.

The above analyses associated with the structural integrity of the reactor coolant pump flywheel have been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

The Extended Power Uprate does not affect system pressures for the systems inside containment such that additional missiles could be generated. The EPU will not result in any system configuration changes inside containment that would impact any existing missile barrier considerations. As such, the existing missile protection measures inside containment remain effective for EPU conditions. (Reference 17)

Reactor Coolant Pump Casing Analysis (ASME Code Case N-481 Analysis)

The ASME Section XI Code, up to and including the 1998 Edition, required a volumetric inspection of the RCP casing welds, and a visual inspection of the pressure boundary components. In lieu of performing the required Section XI internal visual and volumetric inspections of RCP Cast Austenitic Stainless Steel (CASS) casings, a fracture mechanics analysis, supplemented by visual examinations, per the requirements of ASME Code, Case N-481 was performed for the original operating period of 40 years. This analysis is contained in the generic industry WCAP-13045, "Compliance to ASME Code Case N-481 of the Primary Loop Pump Casings of Westinghouse Type Nuclear Steam Supply Systems," and PBNP-specific WCAP-14705, "A Demonstration of Applicability of ASME Code Case N-481 to the Primary Loop Pump Casings of the Point Beach Units 1 and 2." These analyses incorporated the effects of thermal embrittlement, and demonstrated compliance with Code Case N-481 requirements for the original 40-year operating license period.

The current ASME Section XI Code applicable for PBNP does not require pump casing weld volumetric or routine internal visual examinations. Thus, the fracture mechanics analysis is not necessary for the extended period of operation in support of applying Code Case N-481 to eliminate casing volumetric examinations. However, the Generic Technical Report (GTR) for Class 1 Piping and Associated Pressure Boundary Components, WCAP-14575-A, "License Renewal Evaluation: Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components," identifies that a fracture mechanics analysis performed for the extended operating period is an acceptable means of managing thermal aging of CASS. Thus, the Code Case N-481 integrity analysis was evaluated throughout the extended period of operation.

Westinghouse performed an evaluation of the Code Case N-481 integrity analysis to identify if it is acceptable for the extended operating period. The results of the evaluation show that the ASME Code Case N-481 integrity analysis conclusions, documented in WCAP-13045 and WCAP-14705 for the PBNP Units 1 and 2 RCP casings remain valid for the 60-year licensed operating period. An additional evaluation confirmed that the Analysis of Record remain bounding and applicable for EPU conditions (Reference 20).

The Reactor Coolant Pump Integrity Analysis has been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

Reactor Coolant System Main Loop Piping Leak-Before-Break Analysis

In response to Unresolved Safety Issue (USI) A-2 (Asymmetric Blowdown Loads on the Reactor Coolant System), Westinghouse performed a generic Leak-Before-Break (LBB) analysis, which was applicable to PBNP. The LBB analysis was performed to show that any potential leaks that develop in the Reactor Coolant System loop piping can be detected by plant monitoring systems before a postulated crack causing the leak would grow to unstable proportions during the 40-year plant life. The NRC reviewed and approved the generic Westinghouse LBB evaluation in NRC Generic Letter 84-04, “Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Main Loops.” By letter (Reference 3) dated May 6, 1986, the NRC acknowledged that PBNP was bounded by the generic Westinghouse LBB analysis and met the additional criteria identified in NRC Generic Letter 84-04.

A plant-specific LBB analysis for the PBNP Units 1 and 2 primary coolant loop piping was subsequently performed by Westinghouse in 1996, and subsequently revised in 2002 and 2003. The results of the current PBNP LBB analysis are documented in WCAP-14439, Revision 2, “Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Point Beach Nuclear Plant Units 1 and 2 for the Power Uprate and License Renewal Program.” The report demonstrates compliance with LBB technology for the PBNP RCS piping based on plant-specific analysis, using the methodology and criteria of Standard Review Plan Section 3.6.3. The revised LBB analysis incorporates analysis parameters associated with power uprate conditions of up to 10.4 percent reactor power, and a 60-year operating period. This revision documents the plant specific reactor coolant system main loop piping geometry, loading, and material properties used in the fracture mechanics evaluation. Since the primary loop piping systems include cast stainless steel fittings, end of life (60-year) fracture toughness, considering the effects of thermal aging, was determined for each heat of material.

Based on loading, pipe geometry, and fracture toughness considerations, enveloping critical locations were determined at which leak-before-break crack stability evaluations were made. Through wall flaw sizes were found which would cause a leak at a rate of ten times the leakage detection system capability of the plant. Large margins for such flaw sizes were demonstrated against flaw instability. Finally, using the plant specific transients and cycles, fatigue crack growth for the 60 years was shown to be acceptable for the primary loop piping. All the recommended LBB margins (margin on leak rate, margin on flaw size, and margin on loads) were satisfied.

The Reactor Coolant System Main Loop Piping Leak-Before-Break Analysis has been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii). This analysis was also reevaluated to address Extended Power Uprate conditions (Reference 18).

In addition a flaw tolerance analysis (Reference 5) was completed for the (CASS) elbows in the main reactor coolant piping system for Point Beach Units 1 and 2. The conclusion of that analysis was that even with the thermal aging in the susceptible reactor coolant loop CASS piping material for Point Beach Units 1 and 2, the susceptible piping locations have been shown to be tolerant of large flaws for the period of extended operation and operation at EPU conditions.

Pressurizer Surge Line Piping Leak-Before-Break Analysis

Leak-Before-Break (LBB) analysis for the Unit 1 and 2 pressurizer surge line piping was performed in 1998. The results of the analysis are documented in WCAP-15065. The report demonstrates compliance with LBB technology for the PBNP pressurizer surge line piping based on plant specific analysis. Westinghouse revised WCAP-15065 to include the NRC SER approving the LBB analysis for the PBNP Units 1 and 2 pressurizer surge line piping in 2001. This revision is documented in WCAP-15065-P-A, Revision 1, "Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for the Point Beach Units 1 and 2 Nuclear Plants." The pressurizer surge line LBB analysis incorporates analysis parameters associated with original licensed power conditions, and a 40-year operating period. The LBB analysis includes the effects of thermal stratification, as evaluated for the PBNP surge lines in WCAP-13509, "Structural Evaluation of the Point Beach Units 1 and 2 Pressurizer Surge Lines, Considering the Effects of Thermal Stratification." WCAP-15065-P-A documents the plant-specific pressurizer surge line piping geometry, loading and material properties used in the fracture mechanics evaluation. It should be noted that the pressurizer surge line piping does not include cast stainless steel fittings.

The analysis is consistent with the criteria specified in NUREG-1061 Volume 3, utilizing the modified limit load method as specified in draft Standard Review Plan, Section 3.6.3. Based on loading, pipe geometry, and fracture toughness considerations, enveloping critical locations were determined at which leak-before-break crack stability evaluations were made. Through wall flaw sizes were found which would cause a leak at a rate of ten times the leakage detection system capability of the plant. Large margins for such flaw sizes were demonstrated against flaw instability. Finally, using the plant specific transients and cycles, fatigue crack growth for the 40 years was shown to be acceptable for the pressurizer surge line piping. All the recommended LBB margins (margin on leak rate, margin on flaw size, and margin on loads) were satisfied.

The pressurizer surge line LBB analysis was further evaluated to determine the impact of Extended Power Uprate conditions, and a 60-year operating period. The changes in the NSSS design conditions due to power uprate (Reference 18) did not result in any changes to the piping loads used in the analysis. There are no cast pipe fittings contained in the piping system, therefore thermal aging is not an issue for the extended operating period.

Thermal aging of the SS weld material was considered with saturated conditions (fully aged), and thus is valid for the extended period of operation. The transients and cycles for the 60 year operating period are the same as the transients and cycles used in the 40 year operating period analysis. The impacts of changes in NSSS design conditions, and the 60 year operating period were determined to be negligible. The conclusions of the original LBB analysis, contained in WCAP-15065-P-A, remained unchanged.

The pressurizer surge line LBB analysis has been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

Class 1 Accumulator Injection Line Piping Leak-Before-Break Analysis

Leak-Before-Break (LBB) analysis for the PBNP Unit's 1 and 2 accumulator injection line piping was performed in 1998. The scope of the analysis for the accumulator injection lines also includes the residual heat removal (RHR) return line. The results of the analysis are documented in WCAP-15107. The report demonstrates compliance with LBB technology for the PBNP accumulator injection line piping based on plant specific analysis. Westinghouse revised WCAP-15107 to include the NRC SER approving the LBB analysis for the PBNP Units 1 and 2 accumulator injection line piping in 2001. This revision is documented in WCAP-15107-P-A, Revision 1, "Technical Justification for Eliminating Accumulator Lines Rupture as the Structural Design Basis for the Point Beach Units 1 and 2 Nuclear Plants." The accumulator injection line LBB analysis incorporates analysis parameters associated with original licensed power conditions, and a 40-year operating period. WCAP-15107-P-A documents the plant specific accumulator injection line piping geometry, loading, and material properties used in the fracture mechanics evaluation. It should be noted that the accumulator injection line piping does not include cast stainless steel fittings.

The analysis is consistent with the criteria specified in NUREG-1061 Volume 3, utilizing the modified limit load method as specified in draft Standard Review Plan, Section 3.6.3. Based on loading, pipe geometry, and fracture toughness considerations, enveloping critical locations were determined at which leak-before-break crack stability evaluations were made. Through wall flaw sizes were found which would cause a leak at a rate of ten times the leakage detection system capability of the plant. Large margins for such flaw sizes were demonstrated against flaw instability. Finally, using the plant-specific transients and cycles, fatigue crack growth for the 40 years was shown to be acceptable for the accumulator injection line piping. All the recommended LBB margins (margin on leak rate, margin on flaw size, and margin on loads) were satisfied.

The accumulator injection line LBB analysis was further evaluated to determine the impact of Extended Power Uprate conditions (Reference 18), and a 60-year operating period. The changes in the NSSS design conditions due to power uprate did not result in any changes to the piping loads used in the analysis. There are no cast piping fittings contained in the piping system, therefore thermal aging is not an issue for the extended operating period. Thermal aging of the SS weld material was considered with saturated conditions (fully aged), and thus is valid for the extended period of operation. The transients and cycles for the 60-year operating period are the same as the transients and cycles used in the 40-year operating period analysis. The impacts of changes in NSSS design conditions, and the 60-year operating period were determined to be negligible. The conclusions of the original LBB analysis, contained in WCAP-15107-P-A, remained unchanged.

The accumulator injection line LBB analysis has been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

Class 1 RHR Line Piping Leak-Before-Break Analysis

Leak-Before-Break (LBB) analysis for the PBNP Unit's 1 and 2 residual heat removal (RHR) suction line piping was performed in 1998. The results of the analysis are documented in WCAP-15105. The report demonstrates compliance with LBB technology for the PBNP RHR line piping based on plant specific analysis. Westinghouse revised WCAP-15105 to include the

NRC SER approving the LBB analysis for the PBNP Units 1 and 2 RHR line piping in 2001. This revision is documented in WCAP-15105-P-A, Revision 1, "Technical Justification for Eliminating Residual Heat removal (RHR) Lines Rupture as the Structural Design Basis for the Point Beach Units 1 and 2 Nuclear Plants." The RHR line LBB analysis includes the effects of thermal stratification. The RHR line LBB analysis incorporates analysis parameters associated with original licensed power conditions, and a 40-year operating period. WCAP-15105-P-A documents the plant specific RHR line piping geometry, loading, and material properties used in the fracture mechanics evaluation. It should be noted that the RHR line piping does not include cast stainless steel fittings.

The analysis is consistent with the criteria specified in NUREG-1061 Volume 3, utilizing the modified limit load method as specified in draft Standard Review Plan, Section 3.6.3. Based on loading, pipe geometry and fracture toughness considerations, enveloping critical locations were determined at which leak-before-break crack stability evaluations were made. Through wall flaw sizes were found which would cause a leak at a rate of ten times the leakage detection system capability of the plant. Large margins for such flaw sizes were demonstrated against flaw instability. Finally, using the plant specific transients and cycles, fatigue crack growth for the 40 years was shown to be acceptable for the RHR line piping. All the recommended LBB margins (margin on leak rate, margin on flaw size, and margin on loads) were satisfied.

The RHR line LBB analysis was further evaluated to determine the impact of Extended Power Uprate conditions (Reference 18), and a 60-year operating period. The changes in the NSSS design conditions due to power uprate did not result in any changes to the piping loads used in the analysis. There are no cast piping fittings contained in the piping system, therefore thermal aging is not an issue for the extended operating period. Thermal aging of the SS weld material was considered with saturated conditions (fully aged), and thus is valid for the extended period of operation. The transients and cycles for the 60-year operating period are the same as the transients and cycles used in the 40-year operating period analysis. The impacts of changes in NSSS design conditions, and the 60-year operating period were determined to be negligible. The conclusions of the original LBB analysis, contained in WCAP-15105-P-A, remained unchanged.

The RHR line LBB analysis has been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

Reactor Vessel Head Penetration Analysis

The RPV heads were replaced during each unit's respective refueling outage in 2005. All analyses associated with the new RPV heads have been evaluated for operation through EOLE in accordance with 10 CFR 54.21(c)(1)(i). An additional evaluation confirmed that the Analysis of Record remains bounding and applicable for EPU conditions (Reference 20).

15.4.4 LOSS OF PRELOAD

Containment Tendon Loss of Prestress Analysis

The PBNP Units 1 and 2 containment buildings are post-tensioned, reinforced concrete structures composed of vertical cylinder walls and a shallow dome, supported on a conventional reinforced concrete base slab. The cylinder walls and dome are provided with tendons.

The prestress of containment tendons decreases over time as a result of seating of anchorage losses, elastic shortening of concrete, creep of concrete, shrinkage of concrete, relaxation of prestressing steel, and friction losses. New upper limit curves, and lower limit curves, of prestressing forces have been established for all tendons through the period of extended operation. The predicted final effective preload at the end of 60 years exceeds the minimum required preload for all containment tendons. Consequently, the post-tensioning system will continue to perform its intended function throughout the period of extended operation.

The analyses associated with containment tendon loss of prestress have been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

15.4.5 NEUTRON ABSORBER

Spent Fuel Pool Storage Rack Boraflex

The Boraflex Monitoring Program has been discontinued as a result of an NRC approved criticality analysis (Reference 13) that does not credit the presence of Boraflex in the spent fuel pool.

15.4.6 ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT

The NRC has established nuclear plant EQ requirements in 10 CFR 50, Appendix A, Criterion 4, “Environmental and Dynamic Effects Design Bases,” and 10 CFR 50.49. 10 CFR 50.49 specifically requires that an EQ program be established to demonstrate that certain electrical components located in harsh plant environments (that is, those areas of the plant that could be subject to the harsh environmental effects of a loss of coolant accident (LOCA), high energy line break (HELB), or post-LOCA radiation) are qualified to perform their safety function in those harsh environments after the effects of inservice aging. 10 CFR 50.49 requires that the effects of significant aging mechanisms be addressed as part of EQ.

The EQ Program meets the requirements of 10 CFR 50.49 for the applicable electrical components important to safety. 10 CFR 50.49 defines the scope of components to be included, requires the preparation and maintenance of a list of in-scope components, and requires the preparation and maintenance of a qualification file that includes component performance specifications, electrical characteristics and the environmental conditions to which the components could be subjected. 10 CFR 50.49(e)(5) contains provisions for aging that require, in part, consideration of all significant types of aging degradation that can affect component functional capability. 10 CFR 50.49(e) also requires replacement or refurbishment of components qualified for less than the current license term prior to the end of designated life, unless additional life is established through ongoing qualification. 10 CFR 50.49(f) establishes four methods of demonstrating qualification for aging and accident conditions.

10 CFR 50.49(k) and (l) permit different qualification criteria to apply based on plant and component vintage.

The license renewal rule, 10 CFR 54, requires that for each structure and component subject to an Aging Management Review (AMR), the licensee shall demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. The EQ Program manages component thermal, radiation and cyclical aging, as applicable, through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods. As required by 10 CFR 50.49, EQ components not qualified for the current license term are to be refurbished, replaced, or have their qualification extended prior to reaching the aging limits established in the evaluation.

Aging evaluations for EQ components that specify a qualification of at least 40 years are considered time-limited aging analyses (TLAA) for license renewal. The PBNP EQ Program ensures that these EQ components are maintained within the bounds of their qualification bases.

EQ equipment is identified and tabulated in the Master List of Electrical Equipment to be Environmentally Qualified (EQML). This list references the Equipment Qualification Summary Sheets (EQSS), which contain pertinent information that establishes qualified life and applicable environmental parameters.

The EQ Program has been demonstrated to be capable of programmatically managing the qualified lives of the components falling within the scope of the program for license renewal. Based upon a review of the existing program and operating experience, the effective implementation of the EQ Program will provide reasonable assurance that (a) the aging effects will be managed, and (b) EQ components will continue to perform their intended function(s) consistent with the current licensing basis for the period of extended operation.
Therefore, the EQ Program will be an acceptable aging management program for license renewal under 10 CFR 54.21(c)(1)(iii) during the period of extended operation.

The effect of the Extended Power Uprate on environmental conditions inside and outside containment on the qualification of electrical equipment was evaluated. Electrical equipment will continue to meet the relevant requirements of 10 CFR 50.49 following implementation of EPU. (Reference 17)

15.4.7 UNIT 1 PRESSURIZER FLAW EVALUATION

Results of Analysis

The fracture mechanics analysis presented in Calculation PBCH-14Q-301, shows that the current flaw identified during the Unit 1 fall 2005 outage in the Pressurizer upper shell-to-upper head weld is acceptable per the criteria of ASME Section XI, IWB-3612. The calculated maximum stress intensity factor for the observed flaw is 27.99 ksi-inch, as compared to the allowable value of 63.25 ksi-inch, which includes required safety margins as noted in Section 2 of this calculation. In fact, the flaw could grow to more than twice the current size and remain acceptable.

The fatigue growth calculation demonstrates that over more than 200 cycles from 0 to 30 ksi, the resulting flaw growth is insignificant compared to the current size of the flaw. Therefore, growth of the flaw to an unacceptable size over the remaining life of the plant (assumed 60-year operating license) is not predicted.

Degradation Mechanisms

The observed flaw is a subsurface flaw that is remote from any surface (either the wetted inside surface or the air outside surface). Such a flaw is therefore not a result of chemistry-driven mechanisms such as stress corrosion cracking or corrosion. Furthermore, the original fatigue analysis summarized that at this general location in the pressurizer, the cumulative fatigue usage factor over the life of the plant is less than 1.0, so flaw initiation by a fatigue mechanism is not plausible. ASME Section III limits fatigue usage over the life of the plant to less than 1.0, to limit fatigue damage such that fatigue cracks will not initiate. These factors lead to the conclusion that the observed flaw is in fact an artifact of original fabrication, and not to an active degradation mechanism. The evaluation of the hypothetical flaw growth by a fatigue mechanism is, therefore, conservative.

Conclusions and Discussions

Based on the results of the evaluation presented in this calculation package, the indication found during the inservice inspection of the pressurizer welds are acceptable and meet the requirement of ASME Code, Section XI, IWB-3610.

The indication area is about 0.5 in^2. The area of the upper shell to head weld is about 1164 in^2, assuming an inside radius of 42 inches, and a wall thickness of 4.41 inches. The area reduction is less than 0.043% of the original area. This area reduction will have no significant affect on the hoop stress in the weld. Thus, the pressurizer stress analysis based on ASME Boiler and Pressure Vessel Code Section III in Reference 6 is not affected. Therefore, the requirement of IWB-3610 (d) (2) is satisfied.

The post EPU power uprate loading/stresses and conditions remain the same or continue to be conservatively bounded by the assumptions and design input in Reference 6 (Reference 19).

15.4.8 UNIT 1 STEAM GENERATOR B FLAW EVALUATION

Results Of Analysis

The fracture mechanics analysis of a discovered flaw in the Unit 1 Steam Generator B transition cone weld presented in Calculation PBCH-14Q-302 Revision 3, shows that the bounding flaw is acceptable per the criteria of ASME Section XI, IWB-3612. The calculated maximum stress intensity factor for the observed flaw is 42 ksi-inch, as compared to the allowable value of 63.25 ksi-inch, which includes required safety margins as noted in Section 2 of this calculation. In fact, this flaw could grow to slightly more than twice the current size and remain acceptable. All actual flaws are smaller than this assumed bounding flaw.

The fatigue growth calculation demonstrates that over more than 3900 cycles from 0 to 64.7 ksi, the resulting flaw growth of the assumed bounding flaw remains below the allowable flaw size. Most transients experienced by the component are much less severe than this transient, and would lead to negligible growth. Therefore, growth of the flaw to an unacceptable size over the remaining life of the plant (assumed 60-year operating license) is not predicted.

The bounding flaw analyzed in this calculation is much more severe than are any of the flaws in this weld that were accepted under the Acceptance Standards of IWC-3510. Therefore, although fracture mechanics evaluation of such acceptable flaws is not required, the fracture mechanics analysis in this calculation could conservatively be applied to such flaws, if necessary.

Degradation Mechanisms

The observed flaws are subsurface flaws that are remote from any surface (either the wetted inside surface or the air outside surface). Such a flaw is therefore not a result of chemistry-driven mechanisms such as stress corrosion cracking or corrosion. These factors lead to the conclusion that the observed flaws are in fact artifacts of original fabrication, and not due to an active degradation mechanism. The evaluation of the hypothetical flaw growth by a fatigue mechanism is therefore conservative.

Conclusions And Discussions

Based on the results of the evaluation presented in this calculation package, the indications found during the inservice inspection of the Steam Generator B transition cone weld are acceptable and meet the requirement of ASME Code, Section XI, IWB-3610.

The total of all indication areas is about 9.2 in^2. The area of the steam generator weld is about 2012 in^2, assuming a circumference of 524 inches, and a wall thickness of 3.84 inches. The transverse area reduction is less than 0.5% of the original area. This area reduction will have no significant affect on the hoop stress in the weld. Thus, the steam generator stress analysis based on ASME Boiler and Pressure Vessel Code Section III is not affected. Therefore, the requirement of IWB-3610 (d) (2) is satisfied.

The post EPU power uprate loading/stresses and conditions remain the same or continue to be conservatively bounded by the assumptions and design input in Reference 7 (Reference 19).

15.4.9 UNIT 1 STEAM GENERATOR A FLAW EVALUATION

Results of Analysis

The fracture mechanics analysis of a discovered flaw in the Unit 1 Steam Generator A transition cone weld presented in Calculation PBCH-14Q-303 Revision 1, shows that flaw 19 is acceptable per the criteria of ASME Section XI, IWB-3612. The calculated maximum stress intensity factor for the observed flaw is 40 ksi-inch, as compared to the allowable value of 63.25 ksi-inch, which includes required safety margins (10) as noted in Section 2 of this calculation.

The fatigue growth calculation demonstrates that over more than 4800 cycles from 0 to 64.7 ksi, the resulting flaw growth of the flaw remains below the allowable flaw size. Most transients experienced by the component are much less severe than this transient, and would lead to negligible growth. Therefore, growth of the flaw to an unacceptable size over the remaining life of the plant (assumed 60-year operating license) is not predicted.

The flaw analyzed in this calculation is more severe than are any of the flaws in this weld that were accepted under the Acceptance Standards of IWC-3510. Therefore, although fracture mechanics evaluation of such acceptable flaws is not required, the fracture mechanics analysis in this calculation could conservatively be applied to such flaws, if necessary.

Degradation Mechanisms

The observed flaws are subsurface flaws that are remote from any surface (either the wetted inside surface or the air outside surface). Such a flaw is therefore not a result of chemistry-driven mechanisms such as stress corrosion cracking or corrosion. These factors lead to the conclusion that the observed flaws are in fact artifacts of original fabrication, and not due to an active degradation mechanism. The evaluation of the hypothetical flaw growth by a fatigue mechanism is therefore conservative.

Conclusions And Discussions

Based on the results of the evaluation presented in this calculation package, the indications found during the inservice inspection of the Steam Generator A transition cone weld are acceptable and meet the requirement of ASME Code, Section XI, IWB-3610.

The total of all indication areas is about 5.06 in^2. The area of the steam generator weld is about 1928 in^2, assuming a circumference of 524 inches (Reference 3), and a wall thickness of 3.68 inches. The transverse area reduction is less than 0.26% of the original area. This area reduction will have no significant affect on the hoop stress in the weld. Thus, the steam generator stress analysis based on ASME Boiler and Pressure Vessel Code Section III is not affected. Therefore, the requirement of IWB-3610 (d) (2) is satisfied.

The post EPU power uprate loading/stresses and conditions remain the same or continue to be conservatively bounded by the assumptions and design input in Reference 8 (Reference 19).

15.4.10 FLAW TOLERANCE EVALUATION FOR SUSCEPTIBLE CASS REACTOR COOLANT PIPING COMPONENTS IN POINT BEACH UNITS 1 AND 2

The susceptible piping locations in the reactor coolant loop piping system of Point Beach Nuclear Plant Units 1 and 2 were evaluated in accordance with the evaluation procedures and acceptance criteria in Paragraph IWB-3640 of ASME Section XI code. The reactor coolant loop A376 TP316 piping material is not susceptible to thermal aging, but some of the A351 CF8M piping elbow material is susceptible due to the δ-ferrite content level. The maximum acceptable flaw size for a range of flaw shapes for the susceptible CASS piping locations in the hot leg, crossover leg and cold leg are shown in Figures 6-1 to 6-6 of the report. The limiting flaw sizes for a given aspect ratio (R/a) are those shown in Figure 6-1 of the report for longitudinal flaws in the hot leg. From Figure 6-1 of the report, the maximum acceptable initial flaw depth is about 28% through-wall for a flaw with an aspect ratio of 6. Considering the hot leg wall thickness of

2.5 inch, a longitudinal flaw of 0.70 inch in depth and 4.2 inches in length would remain acceptable in accordance with the acceptance criteria of IWB- 3640 for the next 30 years, which represents the remaining plant life for both Point Beach Units 1 and 2 (assumed 60-year operating license). The acceptable initial flaw depths for circumferential flaws and flaws in the crossover leg and cold leg are larger as shown in Table 6-1 of the report. In addition, this maximum acceptable initial flaw depth is deeper than the recommended postulated flaw depth shown in Table L-3210-1 in Article L-3000 of the Code. Therefore, even with thermal aging in the susceptible reactor coolant loop CASS piping material for Point Beach Units 1 and 2, the susceptible piping locations have been shown to be tolerant of large flaws.

The flaw tolerance evaluation of CASS piping material at EPU conditions did not find any significant impact due to thermal aging. (Reference 5)

15.4.11 UNIT 1 REACTOR VESSEL INLET NOZZLE FLAW EVALUATION

Results of Analysis

Phased array ultrasonic examinations of the reactor vessel inlet nozzle-to-pipe weld (RC-32-MRCL-AIII-03) resulted in an American Society of Mechanical Engineers (ASME) Section XI Code rejectable indication in the “A” loop. The weld is a dissimilar metal weld (between the cast stainless elbow and carbon steel nozzle using stainless steel filler material). The indication was recorded 18 inches from top dead center (TDC) and 2.1 inches from the weld centerline on the nozzle side of the weld in the nozzle forging, and approximately 0.9 inches from the buttering. The indication is volumetric in nature (e.g., slag inclusion) and the indication orientation is predominantly circumferential in nature.

Due to inability to meet the ASME Section XI, Appendix VIII, Supplement 10 (dissimilar metal weld) required 0.125 inch root mean square (RMS) acceptance criterion, the flaw was evaluated in accordance with Performance Demonstration Initiative (PDI) policy (PDI 03-01) as allowed by an approved PBNP relief request (RR-21). In addition, because there were no procedurally demonstrated techniques for determining that indications close to the inside surface are, in fact, sub-surface; the indication was treated as surface-connected during the ASME Section XI evaluation(s).

Degradation Mechanisms

Based on current findings, it is considered that this indication or group of indications is most likely to be embedded fabrication flaws; however, it is being evaluated as a surface-connected flaw due to the proximity to the inside surface. The location is actually near to the buttering of the nozzle, and also near to the clad-to-base metal interface.

Conclusions And Discussions

The Section XI Flaw Evaluation of Indication Recorded on RC-32-MRCL-AIII-03 of the Point Beach Unit 1 Inlet Nozzle to Pipe Weld presented in Technical Note LTR-PAFM-10-50-NP shows the flaw is acceptable per Section XI, paragraph IWB-3600. Based on the results of the evaluation presented in the technical note, the indication was found to be acceptable for further service without repair for the remainder of the life of Unit 1, including the period of renewed operation (Reference 14) and operation at EPU conditions (Reference 15). The LBB analysis described in FSAR 15.4.3 remain valid.

15.4.12 REFERENCES

1. NRC Letter, “Acceptance for referencing of Topical Report WCAP-14535-A, “Topical report on Reactor Coolant Pump Flywheel Inspection Elimination,” dated September 12, 1996.

2. NMC Letter to NRC, NRC 2001-059, “Reactor Coolant Pump Flywheel Inspection Interval Change Point Beach Nuclear Plant, Units 1 and 2,” dated September 17, 2001.

3. NRC Letter to WE, “Docket Nos. 50-266 and 50-301,” “Exemption from the Requirements of 10 CFR 50 Appendix A, General Design Criterion 4,” dated May 6, 1986.

4. NUREG/CR-6260 (INEL-95/0045), “Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components,” U.S. Nuclear Regulatory Commission, March 1995.

5. Westinghouse Report LTR-PAFM-05-58, Revision 1. “Flaw Tolerance Evaluation for Susceptible CASS Reactor Coolant Piping Components in Point Beach Units 1 and 2,” dated June 2011.

6. Structural Integrity Associates, Inc. Calculation PBCH-14Q-301 Revision 0, Point Beach Unit 1, “Pressurizer Flaw Evaluation,” October 2005.

7. Structural Integrity Associates, Inc. Calculation PBCH-14Q-302, Revision 3, Point Beach Unit 1, “Steam Generator B Flaw Evaluation,” November 2005.

8. Structural Integrity Associates, Inc. Calculation PBCH-14Q-303, Revision 1, Point Beach Unit 1, “Steam Generator A Flaw Evaluation,” November 2005.

9. Westinghouse Report, CN-REA-08-39, “Neutron Fluence Exposure Evaluations for the Point Beach Units 1 and 2 EPU,” December 2008.

10. AREVA Document BAW-2467P, Revision 1, “Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessel of Point Beach Units 1 and 2 for Extended Life through 53 Effective Full Power Years,” dated October 2004 (Incorporated by reference).

11. NRC Safety Evaluation Related to Amendment Nos. 227/232 to Renewed Facility Operating License Nos. DPR-24 and DPR-27, “Issuance of Amendments Regarding Review of Reactor Vessel Fracture Mechanics Analysis,” dated May 10, 2007.

12. NRC Safety Evaluation, NUREG-1839 “Safety Evaluation Report Related to the License Renewal of the Point Beach Nuclear Plant, Units 1 and 2,” dated December 2005

13. NRC Safety Evaluation, “Point Beach Nuclear Plant, Units 1 and 2 - Issuance of Amendments Re: Spent Fuel Pool Storage Criticality Control,” dated March 5, 2010.

14. NEXTERA Energy letter NRC, NRC 2010-0050, “Unit 1 Refueling 32 Analytical Evaluation Report for the Reactor Vessel Point Beach Nuclear Plant,” dated April 13, 2010.

15. Westinghouse Report LTR-PAFM-10-50-NP, Revision 1, "Section XI Flaw Evaluation of Indication Recorded on RC-32-MRCL-AIII-03 of the Point Beach Unit 1 Inlet Nozzle to Pipe Weld," dated June, 2011.

16. NRC Safety Evaluation, "Point Beach Nuclear Plant, Units 1 and 2 - Issuance of Amendment (Nos. 250 and 254) Regarding Change to Technical Specification 5.6.5, Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR) (TAC Nos. MF0532 and MF0533)," dated June 30, 2014.

17. FPL Energy letter to NRC, NRC 2009-0030, "License Amendment Request 261 Extended Power Uprate," dated April 7, 2009.

18. NRC Safety Evaluation, "Issuance of License Amendment Regarding Extended Power Uprate (TAC Nos. ME1044 and ME1045)," dated May 3, 2011.

19. Structural Integrity Associates, Inc. letter to NextEra Energy, "Reconciliation of Pressurizer and Steam Generator Flaw Evaluation Calculations to Incorporate Extended Power Uprate Conditions," dated September 8, 2011.

20. WCAP-16983-P, Revision 0, "Point Beach Units 1 and 2 Extended Power Uprate (EPU) Engineering Report," (Proprietary) dated September 2009.

21. WCAP-16669-NP, Revision 1 "Point Beach Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation dated January 2009.

15.5 EXEMPTIONS

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