March 30, 2005

Mr. Dennis L. Koehl
Site Vice President
Nuclear Management Company, LLC
6610 Nuclear Road
Two Rivers, WI 54241

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2, LICENSE RENEWAL APPLICATION (TAC NOS. MC2099 AND MC2100)

Dear Mr. Koehl:

By letter dated February 25, 2004, Nuclear Management Company, LLC, (NMC or the applicant) submitted an application pursuant to 10 CFR Part 54, to renew the operating licenses for Point Beach Nuclear Plant (PBNP), Units 1 and 2, for review by the U.S. Nuclear Regulatory Commission (NRC). The NRC staff is reviewing the information contained in the license renewal application (LRA) and has identified, in the enclosure, areas where additional information is needed to complete the review.

These RAIs were discussed with your staff, Mr. Jim Knorr, and a mutually agreeable date for this response is within 30 days from the date of this letter. If you have any questions, please contact me at 301-415-3703 or e-mail VMR1@nrc.gov.

Sincerely,

/RA/

Veronica M. Rodriguez, Project Manager
License Renewal Section A
License Renewal and Environmental Impacts Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket Nos. 50-266 and 50-301

Enclosure: As stated

cc w/encls: See next page

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DISTRIBUTION: Letter to D. Koehl, Re: RAI for review of Point Beach, Units 1 and 2, License Renewal Application, Dated: March 30, 2005
Adams Accession No.: ML050890003

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POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2
LICENSE RENEWAL APPLICATION (LRA)
REQUEST FOR ADDITIONAL INFORMATION (RAI)

Aging Management of Reactor Coolant System

RAI 3.1.1-1

LRA Table 3.1.1 (Page 3-68) states the following:

> "PBNP does not have CASS RCS piping, but does have CASS primary loop elbows. Since the primary loop elbows are cast austenitic stainless steel, the Leak-Before-Break (LBB) analysis for Class 1 piping must consider the effects of thermal embrittlement. The LBB analysis has been identified as a TLAA and is discussed further in Section 4.4. The TLAA was resolved by performing a fracture mechanics evaluation considering loading, pipe geometry, and fracture toughness reduction due to thermal embrittlement to assess LBB crack stability for the period of extended operation. This evaluation demonstrates that a significant margin exists between detectable flaw size and flaw instability. PBNP has chosen the evaluation method to disposition reduction in fracture toughness due to thermal embrittlement of primary loop elbows. Accordingly, an aging management program to manage this effect for the primary loop pipe fittings is not required."

Of particular concern is the statement that:

> "This evaluation demonstrates that a significant margin exists between detectable flaw size and flaw instability. PBNP has chosen the evaluation method to disposition reduction in fracture toughness due to thermal embrittlement of primary loop elbows."

The first sentence implies that the applicant is relying on leakage to detect the presence of a through-wall crack and the existence of the LBB as the means to manage the thermal aging effect for these cast austenitic stainless steel (CASS) elbows. NUREG-1800, Appendix A, Paragraph A.1.2.3.4, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," states that "detection of aging effects should occur before there is a loss of the structure and component intended function." One of the functions associated with these CASS elbows is the pressure boundary. The presence of leakage as permitted by LBB violates the pressure boundary and does not satisfy the NUREG-1800 criteria. Therefore, the staff cannot accept the use of LBB to manage breaching of the elbow pressure boundary resulting from CASS thermal aging.

Furthermore, the staff does not agree that the PBNP decision to use the "evaluation method" satisfies the 10 CFR 54 criteria to manage an aging effect. LBB is a design criteria, not a method of aging management and does not appropriately address the aging effects due to loss of fracture toughness. The Rule, in accordance with 10 CFR 54.21(a)(3), requires that aging effects are managed. CASS thermal aging (loss of fracture toughness) is an aging effect that requires management. The use of LBB is not recognized by the staff as an aging management method and has not been accepted as part of other recent LRA reviews. As such, the staff does not agree with the LRA statement, “Accordingly, an aging management program to manage this effect for the primary loop pipe fittings is not required.”

GALL Report AMP XI.M12, “Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS),” provides acceptable options that the staff believes would be acceptable for managing thermal embrittlement of CASS. These options include “enhanced volumetric examination to detect and size cracks or plant of component-specific flaw tolerance evaluation.”

The staff requests the applicant to clarify if its intended to follow the recommendation of GALL Report XI.M12, otherwise, the applicant is requested to propose an alternate aging effect method that addresses the loss of fracture toughness resulting from the CASS thermal aging effect for these elbows. If an alternative method is to be proposed, the staff requests the applicant to provide a full explanation based on loss fracture toughness as to why the proposed methodology will assure that cracks resulting from the loss of fracture toughness will not breach the pressure boundary of the CASS elbows.

RAI 3.1.1-2

The staff believes that the steam generator (SG) feedrings and associated J-tubes should be included in the scope of license renewal. Since this component is completely enclosed by safety-related pressure-boundary components, it is important to show that failures of this component could not impede certain safety-related functions of the components in which it is contained, as required by 10 CFR 54.4(a)(2).

This is supported by industry experience that demonstrated an aging effect, which if permitted to continue, could result in loose parts and could interfere with a safety component performing its intended function. Examples of the operating experience information compiled by both the Institute of Nuclear Power Operations (INPO) and World Association of Nuclear Operators (WANO) are presented as follows:

- On April 25, 1989, a Westinghouse Model 51 steam generator was undergoing periodic testing and J-nozzles were found to be eroded/corroded.
- On January 14, 1995, a Westinghouse Model 51 steam generator was undergoing a special leak test of the feedring and water appeared to be flowing from a fault in either the piping or plug located at the support bracket, 180 degrees from the feedwater inlet. Two J-nozzles were removed from the top of the feedring, the area was cleaned, and the damaged weld in the bottom of the feedring was weld repaired. Although there was no failure on the J-nozzles, two existing J-nozzles were replaced with new style Inconel J-nozzles as a preventive measure.

The staff requests the applicant to justify why the SG feedrings and associated J-tubes should not be scoped into the license renewal. If such justification is not available, the staff requests the applicant to provide the aging management program that PBNP will be committing to manage these aging effects.

Aging Management of Containments, Structures, and Component Supports

RAI 3.5-12

In LRA Section 3.5.2.2.1.7, the applicant stated that Stress Corrosion Cracking (SCC) is not an applicable aging mechanism for penetration sleeves, bellows, and dissimilar metal welds. The applicant also stated that the PBNP liner penetration have had a fatigue review and are bounded by line Item 3.5.1-01/II.A3.1-b. Therefore, the applicant did not addressed cracking due to cyclic loading (see LRA Section 4.3.11 TLAA - Containment Liner Plate Fatigue Analysis).

SRP-LR Section 3.5.2.2.1.7 identifies that cracking of containment penetrations (including penetration sleeves, penetration bellows, and dissimilar metal welds) due to cyclic loading or SCC could occur in all types of PWR and BWR containments and a visual examination (VT-3) would not detect such cracks. The GALL Report recommends further evaluation of the inspection methods implemented to detect these aging effects. Operating experience of Information Notice (IN) 92-20 describes an instance of containment bellows cracking, resulting in loss of leak tightness.

The staff requests the applicant to address the difference between PBNP's position and the GALL Report recommendation of enhanced inspection methods.

Additionally, the staff noted that the TLAA does not detect and manage cracking due to cyclic loading. Therefore, the staff requests the applicant to provide further justification for crediting Item 3.5.1-01/II.A3.1-b to manage cracking due to cyclic loading.

RAI 3.5-13

(1) In LRA Section 3.5.2.2.1.1, the applicant stated that concrete degradation in air due to aggressive rainwater is insignificant and the below-grade/lake water environment is non-aggressive. The staff requests the applicant to provide sufficient data to support this statement.

(2) During its AMR review, the staff was unable to identify how the LRA addresses the items described in the Interim Staff Guidance (ISG)-03. The staff requests the applicant to identify how its AMRs address all the items described ISG-03.

Flow Accelerated Corrosion Program

RAI B2.1.11-1

GALL XI.M17 states "Inspection results are used as input to a predictive computer code, such as CHECWORKS, to calculate the number of refueling or operating cycles remaining before the component reaches the minimum allowable wall thickness. If calculations indicate that an area will reach the minimum allowed thickness before the next scheduled outage, the component is to be repaired, replaced, or reevaluated." This statement is repeated in the LRA Section B2.1.11, Page B-123.

The PBNP flow accelerated corrosion (FAC) procedures and inspection schedule are developed to provide reasonable assurance that structural integrity will be maintained between inspections. The LRA stated that if the minimum measured thickness is less than 70% of pipe nominal wall thickness, the sample size must be expanded. The acceptance criteria of B2.1.11 states the following:

> "For example, if minimum measured thickness is less than 70% of pipe nominal wall thickness the sample size must be expanded. The expansion must include a minimum of the next two most susceptible components in that CHECWORKS line, any component within two diameters downstream (upstream if expander), or like components in parallel trains. If the initial expansion finds additional components with significant loss of material due to FAC, the examination scope is expanded further. If the measured wall thickness is less than T-min, a local thinning evaluation is performed using the methodology of an approved ASME Section XI Code Case. If the component cannot satisfy the local thinning evaluation, it must be replaced or repaired."

During a telephone conference, dated March 17, 2005, PBNP personnel indicated that the FAC implementation procedures used a T-min setting of 60% of nominal wall thickness, for nonsafety-related SSCs. Safety-related SSCs used another T-min setting.

Based on staff's understanding, the minimum wall is defined in the ASME Code as no less than 87.5% of the nominal wall thickness. The staff is unclear on the basis of how PBNP justifies its T-min wall thicknesses. The staff requests the applicant to provide a detailed description of the methodology used to establish the T-min limits for both safety and non-safety related SSCs.

Furthermore, once the T-min is established, the AMP suggests that the inspection sample will be expanded. The staff is unclear on the basis for the expanded sample and how the applicant will determine the operability of the degraded SSC. The staff requests the applicant to provide an explanation that demonstrates how reasonable assurance is achieved so that structural integrity will be maintained between inspections.

Aging Management Programs

<u>RAI B2.1</u>

Several currently approved relief requests, shown in the attached Table 1, were reviewed by the project team during the audit and review of AMPs B2.1.1, "ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program" and B2.1.2, "ASME Section XI, Subsections IWE and IWL Inservice Inspection Program." The relief requests were presented as the bases for taking exceptions to the following GALL Report AMPs:

(1) GALL Section XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD"
(2) GALL Section XI.M3, "Reactor Head Closure Studs"
(3) GALL Section XI.S1, "ASME Section XI, Subsection IWE"
(4) GALL Section XI.S2, "ASME Section XI, Subsection IWL"

Relief requests are approved by the NRC as described in 10 CFR 50.55a, Codes and Standards. Relief requests only apply to the current licensing basis (CLB) issues and are time limited. Consequently, citing approved relief requests cannot be used as a basis for taking exception to the GALL since they may not be renewed.

Each exception to the GALL must be evaluated for NRC approval based on the technical bases that are associated with aging management regardless of whether there is a current, approved, related relief request. Also, it should be noted that approval of an exception to GALL with respect to a plant's AMP does not mean that a relief request that covers the same issue will be approved during the period of plant life extension. The 10 CFR 50.55a process must still be used for relief request approval. Citing a relief request does not provide an acceptable basis to take an exception to GALL.

The staff requests the applicant to provide the technical bases, as it relates to aging management, without referencing the relief request, for the exceptions taken to AMP B2.1.1, "ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program" and to AMP B2.1.2, "ASME Section XI, Subsections IWE and IWL Inservice Inspection Program."

TABLE 1 - Point Beach Nuclear Plant LRA Relief Requests	
Relief Request No.	**Relief Request Description** LRA Pages B-13 through B-16 states, "The following Relief Requests (RR) have been approved by the NRC and have been incorporated into the ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program"
RR 1	Altering the Date of the Start of the Fourth Inspection Interval
RR 2	Use of Later Code Editions
RR 3	Risk Informed Examination of Class 1 and Class 2 Piping Butt Welds (Code Case N-578 and EPRI TR-112657)
RR 4	Alternate Requirements to Repair and Replacement Documentation Requirements and Inservice Inspection Summary Report Preparation and Submission as Required by IWA-4000 and IWA-6000 (Code Case –532-1)
RR 5	Alternate Requirements for VT-2 Visual Examination of Class 1, 2, and 3 Insulated Pressure-Retaining Bolted Connections (Code Case –533-1)
RR 6	Corrective Action for Leakage Identified at Bolted Connections (Code Case –566-1)
RR 7	Alternate Requirements for VT-2 Visual Examination of Class 1, 2, and 3 Insulated Pressure-Retaining Bolted Connections, Section XI, Division 1 (Code Case –616)
RR 8	Successive Inspections (Code Case –624)
RR 9	Alternative to Welding and Brazing Performance Qualification Requirements
RR 10	Relief from Regenerative Heat Exchanger Examinations
RR 11	Emergency Diesel System VT-2 Examination
RR 12	Request for Alternative to ASME Section XI, Appendix VIII, Supplement 10
RR ERR-1	Elimination of VT-3 examinations of seal and gaskets
RR ERR-5	No Successive Examination of Repairs
RR ERR-6	Elimination of Required Bolt Torque or Tension Tests
RR ERR-7	Elimination of the Need for Venting of Leak Chase Channels During Integrated Leak Rate Tests
RR ERR-9	Allowing the Qualification and Certification of NDE Personnel to a Written Practice in Accordance with SNT-TC1A Instead of CP-189
RR LRR-1	Relaxing the Illumination and Direct Examination Distance Requirements of IWA-2210
RR LRR-2	Allowing a General Visual Inspection of Inaccessible Concrete Surfaces Instead of the VT-3 Examination Required by IWL-2510(a)
RR 1-24 (Unit 1)	Use of ASME Code, Section XI, 1998 Edition with Addenda through 2000
RR-2-30 (Unit 2)	Use of ASME Code, Section XI, 1998 Edition with Addenda through 2000