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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Point Beach Nuclear Plant, Units 1 and 2
Dockets 50-266 and 50-301
License Nos. DPR-24 and DPR-27

Response to Request for Additional Information
Regarding the Point Beach Nuclear Plant License Renewal Application
(TAC Nos. MC2099 and MC2100)

By letter dated February 25, 2004, Nuclear Management Company, LLC (NMC), submitted the Point Beach Nuclear Plant (PBNP) Units 1 and 2 License Renewal Application (LRA). On November 17, 2004, the Nuclear Regulatory Commission (NRC) requested additional information regarding Aging Management Review for the Reactor Coolant System and Pressurizer (LRA Section 3.1.2), the Alloy 600 Aging Management Program (LRA Section 2.1.16), and the Leak Before Break TLAA (LRA Section 4.1.1). The enclosure to this letter contains NMC's response to the staff's questions.

On December 1, 2004, the NRC staff verbally provided additional time for NMC to respond to this request for additional information in order for further clarifications to be provided. The clarifications allowed the PBNP License Renewal project staff to clearly understand the information needed.

Should you have any questions concerning this submittal, please contact Mr. James E. Knorr at (920) 755-6863.

Summary of Commitments

Nuclear Management Company, LLC makes the following commitments:

1. NMC will monitor on-going industry activities related to failure mechanisms for small-bore piping, and will evaluate changes to PBNP inspection activities based on industry recommendations.

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2. NMC will use the interim report "PWR Materials Reliability Project Interim Alloy 600 Safety Assessment for US PWR Plants (MRP-44), Part 1: Alloy 82/182 Pipe Butt Welds," and its final version as part of the basis for the Reactor Coolant System Alloy 600 Inspection Program.

3. NMC will submit the Reactor Coolant System Alloy 600 Inspection Program 24 - 36 months prior to the period of extended operation for staff review and approval to determine if the program demonstrates the ability to manage the effects of aging per 10 CFR 50.54.21(a)(3).

I declare under penalty of perjury that the forgoing is true and correct. Executed on January 25, 2005.

Dennis Koehl

Dennis L. Koehl
Site Vice-President, Point Beach Nuclear Plant
Nuclear Management Company, LLC

Enclosure

cc: Administrator, Region III, USNRC
Project Manager, Point Beach Nuclear Plant, USNRC
Resident Inspector, Point Beach Nuclear Plant, USNRC
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ENCLOSURE

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2 LICENSE RENEWAL APPLICATION

The following information is provided in response to the Nuclear Regulatory Commission (NRC) staff's request for additional information (RAI) regarding License Renewal Application (LRA).

The NRC staff's questions are restated below, with the Nuclear Management Company (NMC) response following.

NRC Question RAI 3.1.2-1 (Aging Management Review for Reactor Coolant System and Pressurizer):

In your LRA, you indicated that WCAP-14575-A and 14574-A was approved by the staff. Please indicate the dates of the approval letters/safety evaluation for the subject WCAPs.

NMC Response:

The approval letter for WCAP-14574, "Aging Management Evaluation for Pressurizers," was dated October 26, 2000. The safety evaluation was an enclosure to the referenced approval letter, and was not dated. The WCAP was subsequently republished as WCAP-14574-A, which included a copy of the NRC safety evaluation.

The approval letter for WCAP-14575, "Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components," was dated November 8, 2000. The safety evaluation was an enclosure to the referenced approval letter, and was not dated. The WCAP was subsequently republished as WCAP-14575-A, which included a copy of the NRC safety evaluation.

NRC Question RAI 3.1.2-2 (Aging Management Review for Reactor Coolant System and Pressurizer):

In your LRA application, you indicated that meaningful volumetric inspection techniques did not exist for socket welds in the Class 1 piping. In light of the successful application of UT of socket welded joints in one inch piping at Susquehanna Steam Electric Station, please discuss the applicability for this technique at PBNP Units 1 and 2 to manage the aging effects of fatigue. If you determine that this application is not suitable for your plant, the discussion should describe in detail why this technique is not viable at PBNP Units 1 and 2, and the basis why no safety significant condition exists if not implemented.

NMC Response:

The Susquehanna Steam Electric Station enhanced the Diablo Canyon Ultrasonic Testing (UT) inspection technique for 3/4 and 1-inch socket welds. The "Susquehanna" UT technique was intended to inspect socket welds for early detection of Inside Diameter (ID) fatigue cracking to eliminate unscheduled shutdown costs associated with repair of socket weld failures. The "Susquehanna" UT technique has been applied at the Susquehanna Steam Electric Station to identify questionable socket welds, allow repair, and thus reduce vibrational induced fatigue failures of socket welded piping connections.

Only a limited amount of follow up destructive examinations were performed on questionable socket welds identified by the "Susquehanna" UT technique. Two follow up destructive examinations of questionable welds verified a through-wall crack, and a 40% through-wall crack.

The "Susquehanna" UT technique is only considered a go / no go inspection when an indication is discovered. Characterization of the indication is not possible. The "Susquehanna" UT technique can detect 25 percent through-wall defects with a high probability. The "Susquehanna" UT technique can also detect much smaller defects, however the technical limits have not been determined.

The "Susquehanna" UT technique shows promise in situations dealing with frequent fatigue-related socket welded joint failures. The technique is not a Code acceptable inspection, as it does not meet the inspection requirements of ASME for a volumetric examination. The "Susquehanna" UT technique has not been thoroughly investigated, nor accepted by the industry. In addition, since the "Susquehanna" UT technique cannot characterize indications, application of the technique can result in rejecting socket welds that may otherwise be acceptable, as demonstrated by application of the technique at the Susquehanna Steam Electric Station.

PBNP has not experienced a high number of socket weld fatigue failures in Class 1 piping. The failures to date have been primarily associated with isolated original installation defects or system design deficiencies.

Three (3) 2-inch socket weld cracks were experienced in the auxiliary feedwater pump recirculation lines downstream of the pressure reducing orifices, following a system modification to allow an increase in the pump recirculation flow rates. The cracks were attributed to vibration induced fatigue originating from the pressure reducing orifices. The welds were repaired and the pressure reducing orifices were replaced to eliminate the source of vibration. Each of the cracks resulted in minor leakage that was detected by plant personnel during area inspections.

One (1) 3/4-inch socket weld crack was experienced in a drain line connection in the Safety Injection / Containment Spray systems full-flow test line. The full-flow test line was a system modification to accommodate minimizing operation of safety grade pumps at minimum flow rates. The failure was attributed to vibration induced fatigue originating from the pressure reducing orifice. The weld was repaired and the piping system support scheme was modified to reduce the vibration. The crack resulted in minor leakage. Since the line is normally isolated from the noted systems, the leakage was detected by operations personnel during the performance of the full-flow system test.

Two (2) 3/8-inch socket weld cracks were experienced on the Steam Generator (S/G) channel head drain line isolation valve. The S/G channel head drain lines were included with the replacement Unit 1 S/Gs. The cracks were attributed to fatigue originating from either thermal cycling or the acoustic behavior of the specific geometry and flow conditions. The welds were repaired and increased in size per EPRI guidance. The cracks did not result in significant leakage. The leakage was identified during area inspections.

The small bore piping locations selected by the PBNP Risk-Informed Inservice Inspection (RI-ISI) program are in primary systems, and are located within the containment building. Application of the "Susquehanna" UT inspection technique for socket-welds can erroneously suggest the need for replacement of acceptable socket-welded joints in a radiation environment.

The PBNP RI-ISI program was created based on the guidance of EPRI TR-112657, Revision B-A, "Revised Risk-Informed Inservice Inspection Evaluation Procedure." The EPRI TR-112657 RI-ISI methodology specifically considers a variety of aging degradation mechanisms, including fatigue. Both the PBNP RI-ISI program and EPRI TR-112657, Revision B-A, were reviewed and approved by the NRC.

EPRI TR-112657, Revision B-A, notes the following:

> "The real measure of protection against catastrophic failure of a piping system component is the combination of good design and leak-before-break properties. All of the service induced failure mechanisms which effect nuclear power plant piping except one (flow accelerated corrosion) have been shown to be of a gradually progressing nature, which inevitably produce detectable leakage before significantly reducing the inherent safety margins of the piping relative to gross rupture. The combination of periodic leak tests required by Section XI, in conjunction with continuous leakage monitoring requirements for all primary coolant systems during operation has proven to be more than adequate protection against a large pipe break. The potential for flow accelerated corrosion, which has caused large pipe breaks without prior leakage, is minimal in Class 1 systems."

In addition, the NRC Safety Evaluation of the PBNP RI-ISI program, dated July 2, 2003, notes the following:

> "The objective of ISI required by ASME Code, Section XI, is to identify conditions (i.e., flaw indications) that are precursors to leaks and ruptures in the pressure boundary that may impact plant safety. The RI-ISI program is judged to meet this objective."

"The methodology used by the licensee also considers implementation and performance monitoring strategies. Inspection strategies ensure that failure mechanisms of concern have been addressed and there is adequate assurance of detecting damage before structural integrity is affected. The risk significance of piping segments is taken into account in defining the inspection scope for the RI-ISI program."

In view of the facts that PBNP has only experienced a limited amount of socket weld fatigue failures primarily related to design issues, the "Susquehanna" UT technique can lead to unnecessary radiation exposure in repairing good socket weld joints, industry experience indicates that fatigue degradation will lead to detectable leakage well before a loss of structural integrity, and the existing RI-ISI program has been found adequate to detect damage before structural integrity is affected, NMC does not believe that performing the "Susquehanna" UT inspection technique on socket welds at PBNP is justified. Therefore, NMC concludes no safety significant conditions exist.

The NMC will actively participate in the EPRI sponsored Materials Reliability Project (MRP) Industry Task Group (ITG) on thermal fatigue. In addition, NMC commits to monitor on-going industry activities related to failure mechanisms for small-bore piping, and will evaluate changes to PBNP inspection activities based on industry recommendations.

NRC Question RAI 3.1.2-3 (Aging Management Review for Reactor Coolant System and Pressurizer):

In your plant specific response, item (10), Table 3.1.0-1, you indicated that plant process control procedures (design control, repair/replacement, and welding) will be revised to ensure that repair or replacement of Class 1 piping components welded connections or cast austenitic stainless steel (CASS) would require a new LBB analysis based on replacement process and/or material properties. Prior to that statement, you indicated that the subject LBB analyses had been revised addressing SG replacement, power uprate, and a 60-year operating period. Since these LBB analyses revisions typically address the effects of thermal aging on CASS components, please explain in detail why the revisions to the procedures are necessary. In your response, please advise if the revised LBB analyses address the effects of thermal aging on CASS components.

Secondly, if the revisions to the plant control procedures are required per your plant administrative program controls, please provide a commitment to your LRA that the subject revisions will be completed prior to the period of extended operation.

NMC Response:

The PBNP Leak-Before-Break (LBB) analyses have been revised to address Steam Generator Replacement, Power Uprate, and a 60-year operating period. The revised LBB analyses account for the effects of thermal aging of CASS materials.

Applicant Action Item 10 of the NRC Safety Evaluation for WCAP-14575, "Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components," requires that provisions for management of Thermal Aging (LBB Analyses) include addressing the impacts of repairs/replacements of CASS components on the LBB analyses. This is the reason the proposed revisions to the noted PBNP procedures/processes were deemed necessary.

In NMC Letter to the NRC, NRC 2004-0016, dated February 25, 2004, PBNP has committed to implement an enhanced ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program prior to the period of extended operation. The LRA, Appendix B, Subsection B2.1.1 identifies the necessary program enhancements. The need to revise noted procedures/processes to address maintenance of the LBB analyses is included in the referenced LRA section. Thus, PBNP has already committed to performing the subject enhancements.

NRC Question RAI 3.1.2-4 (Aging Management Review for Reactor Coolant System and Pressurizer):

Under the plant-specific response in Table 3.1.0-3 for Renewal Applicant Action Item (4), you stated that absolute assurance could not be provided that the yield strength of your SA-193, Grade B7 bolting is under 150 ksi. Furthermore, you stated that since the Inservice Inspection database results show that no cracking is occurring, you do not consider SCC an aging effect requiring management for the Point Beach pressurizer bolting. Please explain in detail, the type, extent, and frequency of nondestructive examinations on this pressure retaining bolting performed at Point Beach under your Inservice Inspection Program.

NMC Response:

The pressurizer bolting is inspected in accordance with ASME Section XI, Table IWB-2500-1, Category B-G-2, Item No. B7.20. The inspection is a VT-1 (surface visual) of 100% of the bolting performed once per interval.

In addition to the ASME Section XI bolting inspections, the pressurizer bolting is visually inspected by trained maintenance personnel each time the bolting is removed for maintenance activities. These additional inspections are performed as a result of PBNP's response to NRC IEB 82-02, "Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants." Since the pressurizer bolting at PBNP is removed, cleaned, and re-torqued on a refueling basis (~18 months), the maintenance inspection is also performed on a refueling basis.

NRC Question RAI 3.1.2-5 (Aging Management Review for Reactor Coolant System and Pressurizer):

In Tables 3.1.2-1 and 3.1.2-4, you indicate that the Water Chemistry Control AMP is used to manage the effects of loss of material due to corrosion in low flow and stagnant areas for a variety of stainless and cast stainless materials. These components are designated by Note H with footnotes 5 and 21. NUREG-1801, XI.M2 recommends a One-Time Inspection Program to validate the effectiveness of the Chemistry Control Program for low flow and stagnant areas because the mitigating effects of a Water Chemistry Control program are effective for intermediate and high-flow areas. Please discuss how the aging effect of loss of material for components specified under Note H of the two listed tables is managed for low flow/stagnant areas since NUREG-1801, XI.M2 specifies that the mitigating effects of a Chemistry Control Program alone is not sufficient.

NMC Response:

In NUREG-1801, Chapter IV, Section C2, "Reactor Coolant System and Connected Lines (Pressurized Water Reactor)," none of the line items corresponding to the internal surfaces of Class 1 piping components or the pressurizer, refer to loss of material due to corrosion, they only refer to cracking. (This condition is reflected by the use of Note H, which is simply intended to communicate that the aging effect of "Loss of material" was not included in the GALL tables for these components.) The reason GALL did not identify this aging effect may be because loss of material due to corrosion was considered to be non-significant/not susceptible in a PWR primary coolant environment.* However, this is due to the fact that the PWR primary coolant environment is strictly controlled by the Water Chemistry Program, and therefore PBNP conservatively included this aging effect in Tables 3.1.2-1 and 3.1.2-4 to be managed by the Water Chemistry Program. Note that every component identified with this aging effect also identifies cracking as a separate aging effect that is managed in-part by the Inservice Inspection Program. While these inservice inspections will look primarily for cracking, they can identify loss of material due to corrosion also. Plant-specific and industry operating experience has shown the loss of material due to corrosion is not actively occurring in PWR primary coolant environments. This position is based on many internal inspections that have been (and will continue to be) performed on Class 1 piping and components.

For these reasons, additional one-time inspections are not warranted for the management of loss of material due to corrosion, for components in PBNP's Class 1 piping/components and pressurizer.

* (See NRC FSER for WCAP-14574, "Aging Management Evaluation for Pressurizers," p. 21-22, along with Applicant Action Item 3.2.2.1-1; and NRC FSER for WCAP-14575, "Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components," p. 17.)

NRC Question RAI 2.1.16-1 (Alloy 600 Program):

Please provide a commitment to assure that interim report "PWR Materials Reliability Project Interim Alloy 600 Safety Assessment for US PWR Plants (MRP-44), Part 1: Alloy 82/182 Pipe Butt Welds," and it's final version will be used as part of the basis for the Reactor Coolant System Alloy 600 Inspection Program. The commitment should state that the Reactor Coolant System Alloy 600 Inspection Program will be submitted 24 - 36 months prior to the period of extended operation for staff review and approval to determine if the program demonstrates the ability to manage the effects of aging per 10 CFR 50.54.21(a)(3).

NMC Response:

NMC commits to use the interim report "PWR Materials Reliability Project Interim Alloy 600 Safety Assessment for US PWR Plants (MRP-44), Part 1: Alloy 82/182 Pipe Butt Welds," and its final version as part of the basis for the Reactor Coolant System Alloy 600 Inspection Program. NMC further commits that the Reactor Coolant System Alloy 600 Inspection Program will be submitted 24 - 36 months prior to the period of extended operation for staff review and approval to determine if the program demonstrates the ability to manage the effects of aging per 10 CFR 50.54.21(a)(3).

NRC Question RAI 2.1.16-2 (Alloy 600 Program):

Please discuss in detail your review of industry/plant operating experience and how it will equate to the continued operation of the existing PBNP Units 1 and 2 RPV heads. If the heads are to be replaced, please discuss your plans for the monitoring of the heads in accordance with current industry guidance, Owner's Groups activities and existing NRC regulations or Orders.

NMC Response:

The PBNP RPV Heads will be replaced during each Unit's upcoming refueling outage in 2005.

The replacement PBNP RPV Heads will be inspected in accordance with the requirements of NRC Order EA-03-009, "Issuance of First Revised NRC Order Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors," revised February 20, 2004.

NMC is actively participating with the industry through the EPRI MRP efforts to develop long term inspection requirements for reactor vessel closure heads and their penetrations for U.S. pressurized water reactor plants.

NRC Question RAI 4.1.1-1 (Leak Before Break TLAA):

Please discuss whether there are any calculations or analyses at PBNP that address the topics listed in 4.1.1.1 of the application and were not included in Table 4.1-2 of the LRA.

NMC Response:

There are no calculations or analyses at PBNP that address the topics listed in 4.1.1.1 of the application that were not included in Table 4.1-2 of the LRA.

NRC Question RAI 4.1.1-2 (Leak Before Break TLAA):

In section 4.1.2, pursuant to 10 CFR 54.21(c)(2), a list of plant specific exemptions granted pursuant to 10 CFR 50.12 was provided. This section described the exemptions and why they were still needed. Pursuant to 10 CFR 54.21(c)(2), please provide an evaluation that justifies the continuation of these exemptions for the period of extended operation.

NMC Response:

NMC withdrew the request for exemptions to 10 CFR 50.61, and Appendices G and H to 10 CFR 50 (Reference Letter from NMC to NRC dated August 3, 2004, NRC 2004-0079). Revised sections of the LRA were included in NMC Letter to the NRC dated September 10, 2004 (NRC 2004-0085). The revised Section 4.1.2 now states, "No TLAA related exemptions granted pursuant to 10 CFR 50.12 were identified."