NMC
Committed to Nuclear Excellence

Point Beach Nuclear Plant
Operated by Nuclear Management Company, LLC

March 4, 2005

NRC 2005-0024
10 CFR 54

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 February 7, 2005, the Nuclear Regulatory Commission (NRC) requested additional information regarding the Reactor Coolant Pump (RCP) Casing Time Limited Aging Analysis (LRA Section 4.4.3) and the Bolting Integrity Program (LRA Section B2.1.4). The enclosure to this letter contains NMC's response to the staff's questions.

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

This letter contains no new commitments and no revisions to existing commitments.

I declare under penalty of perjury that the forgoing is true and correct. Executed on March 4, 2005.

Dennis L. Koehl

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

Enclosure

A093

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

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 the Point Beach Nuclear Plant (PBNP) License Renewal Application (LRA).

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

Reactor Coolant Pump (RCP) Casing Analysis

NRC Question RAI 4.4.3-1:

Section 4.4.3 of the application indicates that the applicant has re-evaluated the fracture mechanics analyses to ASME Code Case-481 documented in WCAP-13045 and WCAP-14705 for the Point Beach Nuclear Plant (PBNP) Units 1 and 2 RCP casings and they remain valid for the 60-year extended license operating period.
The application indicates that these components are not susceptible to thermal aging because they satisfy the criteria in the NRC safety evaluation for WCAP-14575-A.
The application also indicates that the fracture mechanics analysis will not be revised and resubmitted to the NRC for the extended period of operation because the code case has been superceded by the ASME Code and the analysis is no longer needed.
The staff requests that the applicant evaluate the ASME Code Case-481 analysis to the criteria for time-limited aging analysis (TLAA) in 10 CFR 54.3 to determine whether the analysis satisfies the criteria and should be considered a TLAA. If it satisfies the TLAA criteria, the applicant is requested to identify the changes to the analysis that result from the proposed additional 20 years of facility operation and to provide the results of the analysis that satisfy 10 CFR 54.21(c)(i), (ii) or (iii).

NMC Response:

The reactor coolant pump integrity analysis is not a time-limited aging analyses (TLAA) for PBNP. As defined in 10 CFR 54.3(a):

> Time-limited aging analyses, for the purposes of this part, are those licensee calculations and analyses that:
>
> (1) Involve systems, structures, and components within the scope of license renewal, as delineated in §54.4(a);
>
> (2) Consider the effects of aging;

(3) Involve time-limited assumptions defined by the current operating term, for example, 40 years;

(4) Were determined to be relevant by the licensee in making a safety determination;

(5) Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, and component to perform its intended functions, as delineated in §54.4(b); and

(6) Are contained or incorporated by reference in the current licensing basis (CLB).

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, and the PBNP specific WCAP-14705. 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 is the 1998 Edition of the code with Addenda through 2000. The NRC approved the use of this ASME Code Edition at PBNP with an NRC safety evaluation dated November 6, 2001. This code does not require pump casing weld volumetric, or routine internal visual examinations; however, it does require external surface examinations of the casing welds and internal visual examinations when the RCP is disassembled for other reasons. Since RCP volumetric examinations are no longer required by the ASME Section XI Code, the fracture mechanics analysis providing the basis for invoking Code Case N-481 is no longer needed and is no longer a part of the PBNP current licensing basis (CLB). Thus, the fracture mechanics analysis for applying Code Case N-481 does not meet the six criteria for defining a TLAA per 10 CFR 54.3 and is not a TLAA for PBNP.

The generic technical report (GTR) for Class 1 Piping and Associated Pressure Boundary Components, WCAP-14575-A, identifies that a fracture mechanics analysis performed for the extended operating period is an acceptable means of managing thermal aging of CASS. The NRC SER for the GTR for Class 1 Piping and Associated Pressure Boundary Components, WCAP-14575-A, provides delta-ferrite and Molybdenum screening criteria to determine if the CASS material is susceptible to thermal aging. The generic industry WCAP-13045 identifies that the delta-ferrite content for all the PBNP Unit's RCP casing castings is less than 10%, and the Molybdenum content is 0.20 percent. The PBNP pump casing castings are not considered susceptible to thermal aging per the screening criteria. Thus, a RCP's structural integrity analysis for managing thermal aging at PBNP is not necessary.

As a result of the above information and at the request of the NRC staff, Section 4.4.3, "Reactor Coolant Pump Casing Analysis (ASME Code Case N-481 Analysis)," is being deleted from the PBNP LRA.

Bolting Integrity Program

NRC Question RAI B.2.1.4-1:

In the Preventive Actions portion of the Bolting Integrity Program, the applicant indicates that the program takes exception to Generic Aging Lessons Learned (GALL) Program XI.M18, "Bolting Integrity." As a preventive action GALL indicates that Initial Inservice Inspection (ISI) of bolting for pressure retaining components should include a check of the bolt torque and uniformity of the gasket compression after assembly. The applicant indicates that these parameters may be checked as part of maintenance activities, but the initial ISI would only include an inspection for leakage of reactor coolant system (RCS) components. The staff requests that the applicant describe the maintenance procedures that are utilized to check bolt torque and uniformity of gasket compression. In addition, please identify the frequency of the maintenance activity.

NMC Response:

NUREG-1801 incorrectly states that bolt torque and uniformity of gasket compression is checked as part of the initial inservice inspection (ISI). The initial ISI requirements only include an inspection for leakage of Reactor Coolant System components. Bolt torque and uniformity of gasket compression are performed as part of the maintenance process and are controlled as part of a formal work control document. Maintenance Instruction MI-32.1 provides generic guidance regarding bolt torque values and uniform gasket compression on typical bolted joints and is used in the development of the work control package if requirements are not otherwise specified by drawings or equipment technical information. Note that the majority of bolted joints at PBNP are designed to ensure the uniformity of gasket compression by the use of metal-to-metal joints, either through the use of a gasket crush ring or a gasket recess in the joint flange. The frequency of a bolted joint maintenance activity is dependant on the need to disassemble and reassemble the joint as part of a corrective maintenance activity of the joint itself or to support corrective or periodic maintenance of associated components.

NRC Question RAI B.2.1.4-2:

In the Preventive Actions portion of the Bolting Integrity Program, the applicant indicates that the program takes exception to GALL Program XI.M18, "Bolting Integrity." GALL Program XI.M18, "Bolting Integrity" indicates that the program relies on recommendations for a comprehensive bolting integrity program, as delineated in NUREG-1339, and industry recommendations, as delineated in the Electric Power

Research Institute (EPRI) NP-5769, with the exceptions noted in NUREG-1339 for safety related bolting. The GALL program relies on industry recommendations for comprehensive bolting maintenance, as delineated in the EPRI TR-104213 for pressure retaining bolting and structural bolting. The applicant indicates that enhancements to the existing plant implementation documents dealing with bolted joints will be made to incorporate recommendations *as deemed appropriate* based upon review of NUREG-1339, EPRI NP-5769, and EPRI TR-104213. The staff should be informed of, and approve, specific exceptions to the bolting recommendations in these NUREG and EPRI documents. The applicant may provide this information either prior to issuance of the extended license or after issuance of the license, provided the information is submitted for review and approval at least two years prior to entering the renewal period. If the information is to be provided after issuance of the extended license, the staff requests the applicant to include this as a commitment.

NMC Response:

As noted in the Scope portion of the Bolting Integrity Program in Section B2.1.4 of the LRA (Page B-54), it is NMC's intention to review the referenced documents and incorporate applicable recommendations into PBNP's bolting practices using the discretionary guidance provided in the referenced documents. Appropriate implementing documents will be revised to include consideration of the referenced document's recommendations when making any changes to the Bolting Integrity Program or when additional guidance, not currently available in the program implementing documents, is needed. Many portions of these referenced documents either do not apply to PBNP, do not make specific recommendations, or may conflict with equipment design and/or engineering specifications. Therefore, it would be inappropriate to characterize PBNP's Bolting Integrity Program as incorporating all aspects of the referenced documents. NMC conservatively considers this an exception to the NUREG-1801 program in order to avoid any potential misunderstanding regarding NMC's intended implementation of the program. NMC's implementation of the Bolting Integrity Program is consistent with the manner in which the NRC characterized the expected use of EPRI NP-5769 and NUREG-1339 in Generic Letter 91-17.

NRC Question RAI B.2.1.4-3:

In the Parameters Monitored or Inspected, Detection of Aging Effects and Acceptance Criteria portions of the Bolting Integrity Program, the applicant identified that the inspection program for high strength (≥150 ksi yield strength) component support bolting associated with the Steam Generator, Reactor Coolant Pump and Reactor Vessel supports would be inspected and tested in accordance with American Society for Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requirements; but, would not be inspected and tested in accordance with the recommendations in GALL Program XI.M18 for high strength bolts. The applicant indicated that these bolts are not susceptible to stress corrosion cracking (SCC) because they are loaded only in shear, they have no preload and are not located in an aggressive environment.

To justify that these bolts are not susceptible to SCC and the additional inspection and test recommendations in GALL Program XI.M18, the staff requests the applicant to either:

a) provide data that demonstrates that the bolting with the maximum certified yield strength, loaded to the maximum shear stress, and in a containment environment would not be susceptible to SCC, or

b) identify the inspection history for these bolts that demonstrates that they are not susceptible to SCC. The inspection history should including the inspection method, date of inspection and the inspection results.

NMC Response:

Cracking of high strength structural bolting materials has been attributed to SCC. Common features of the failures by SCC were that high strength or overly hard materials were used in moist environments under high sustained tensile stresses.

Three parameters are required for stress corrosion cracking to occur: (1) a corrosive environment, (2) a susceptible material, and (3) high sustained tensile stresses. SCC is a phenomenon that primarily occurs in stainless steels and not in carbon steels. Structural carbon steel bolting that has experienced SCC has been limited to materials that are high nickel maraging steels or low-alloy quenched and tempered (LAQT) steels. Structural steel bolting failures due to SCC have been limited to high strength or ultra high strength materials, (i.e., yield strength S_y >150 ksi). The NRC has stated that categorization should be based only on the actual measured yield strength, S_y, of the material (or S_y determined by conversion of measured hardness values) and not the specified minimum yield strength. Note that this discussion concerns the material property of the steel and not the stress of the bolt due to an applied load.

Structural bolted connections for supports of Class 1 components at PBNP are designed and assembled in accordance with the American Institute of Steel Construction (AISC) specifications and use ASTM bolt material A325 or A490. The important mechanical properties of these fasteners are strength, ductility, and resistance to SCC. A490 bolting material has a specified minimum yield strength of 130 ksi, placing it in the medium strength bolt category. Actual yield strength could be in excess of 150 ksi; therefore, A490 bolting should be reviewed for susceptibility to SCC.

Generally in structural design, connections are *friction-type* that require high bolt preloads to clamp the pieces together. A490 bolts are preloaded to a minimum of 70% of the ultimate tensile strength. The bolts will likely be loaded above the yield point at assembly. It is for this reason that A490 bolting is not reused under any circumstance. The other AISC connection design is a *bearing-type*. Bearing joints do not require high bolt preloads. The bolts carry the load in shear and bear across the thickness of the joint plies.

A review of plant drawings and specifications for the supports of Class 1 components was performed to determine the design of the connection and bolt preload requirements. A490 bolting/material is used in two types of applications for the supports of Class 1 components. It functions as a bolt in bolted connections and as a pin in pinned connections (*bearing-type*). The PBNP support drawings detail the use of jam nuts in combination with the bolt nuts for many of the pins and embedded anchor bolts. The use of jam nuts clearly signify that preload was to be excluded. The conclusion drawn is that bolt preload was not intended and that the bolts were not placed in a state of high tensile stress. As a point of reference, the AISC allowable working stress for A490 fasteners is 54 ksi in tension and 32 ksi in shear (threads excluded from shear plane). The bolted connections were most likely tightened (preloaded) in accordance with AISC specifications.

Contaminants such as sulfates, fluorides, or chlorides can provide the necessary environment for SCC. Materials respond differently to environmental (aqueous mediums) and stress conditions. The general environment in a reactor containment building is closely controlled to exclude sulfate, fluoride, or chloride contaminants. Moist environments and effects of boric acid corrosion are avoided. In the past SCC in the presence of bolt lubricants has been a problem. Molybdenum disulfide (MoS_2) has experienced decomposition, resulting in the introduction of H_2S contaminants needed to promote SCC failure. NMC takes care in the selection of lubricants to preclude lubricant-induced SCC.

Plant-specific operating experience history for the supports of Class 1 components is also important information. Currently, 1998 Edition with Addenda through 2000 of ASME Section XI is required to be implemented at PBNP. Under Examination Category F-A, item number F1.40, the bolting on these component supports are examined periodically. The inspection method is VT-3. Personnel are trained and qualified in accordance with ANSI-ASNT CP-189 (IWA-2310). This qualification requires written tests to prove the examiner has the knowledge required for examinations and a practical test to ensure they understand the components they are examining.

The requirement is to examine 100% of the supports. For components other than piping, within a system of similar design, function, and service, the supports of only one of the multiple components are required to be examined. This would mean the supports of one RCP and one steam generator, and the reactor pressure vessel (RPV), would be examined each ten-year interval. The examination requirement includes all bolted connections to the component, to the building structure, and any intervening elements. The acceptance standards (IWF-3410) state that any deformations or structural degradations of fasteners, springs, clamps, or other support items, and missing, detached, or loosened support items are unacceptable for continued service until repaired, replaced, or accepted by evaluation or test. If the acceptance criteria cannot be met, then additional examinations (IWF-2430) shall be performed. This would include the immediately adjacent component supports and additional supports within the system, equal in number and of the same type and function as those scheduled for examination during the inspection period (periods usually include two outages). This would require PBNP to look at the supports on the other RCP and steam generator.

The Boric Acid Program also takes a critical look at bolting. Whenever boric acid is found, the requirement is to look at the flow path of where the boric acid has traveled. If boric acid is found on bolting, the boric acid will be removed and a visual examination performed on the fasteners to determine if any degradation has occurred. NMC will follow plant procedures for repair or replacement if the evaluation determines the bolting is not acceptable.

The inspection history results are:

Component	Unit 1	Unit 2
RCP A	4/20/1992 - NRI 4/18/1994 - NRI 2/27/1998 - NRI each leg examined separately	9/30/1994 - NRI 10/18/1996 - NRI 10/26/2000 - NRI each leg examined separately
RCP B	Note (2)	Note (2)
Steam Generator A	4/30/1991 - NRI 4/20/1992 - NRI 4/8/1994 - NRI 4/27/1998 - NRI 4/5/2004 - NRI	12/18/1996 - NRI 4/16/2002–Note (1)
Steam Generator B	Note (2)	12/18/1996 - NRI
RPV	Note (2)	Note (2)

NRI = No Recordable Indications

(1) Support baseplate to floor was noted as being out of alignment. An evaluation showed this was a construction issue and was initially placed in service in this condition. Accepted as-is.

(2) Records of examinations for these components prior to 1990 are not included. It is assumed examinations have been performed because previous editions of ASME Section XI required these examinations; however, no examination data has been located at this time.

In conclusion, SCC is not an aging effect for the A490 bolting used in the supports of Class 1 components at PBNP. This is based on the totality of the information presented, including plant operating experience. High strength structural bolt failure by SCC requires a combination of high tensile stress and environmental factors. High tensile stresses greater than or equal to the yield strength of the material are required to induce SCC. The bolting and pin joints used in the major supports of Class 1 components at PBNP are not subjected to high preload stresses. The tight controls placed on the exclusion of contaminants in containments and the monitoring for boric acid corrosion or moist environments limits the possibility of a harsh environment.

Finally, operating experience at the plant has not identified any failure of a bolt associated with the supports of Class 1 components.

NRC Question RAI B.2.1.4-4:

In the Monitoring and Trending portion of the Bolting Integrity Program, the applicant indicates that the frequency of inspection of leaking pressure retaining components (not covered by ASME Section XI) will be in accordance with the plant maintenance and/or corrective action process. GALL Program XI.M18 indicates that these components should be inspected daily. If they have leaks and if the leak rate does not increase, the inspection frequency may be decreased to weekly or biweekly. The staff requests that the applicant provide justification for utilizing the plant maintenance and/or corrective action process for determining the inspection frequency for leaking pressure retaining non-ASME Section XI components. The staff requests the applicant to identify how the plant maintenance and/or corrective action process determines the frequency of inspection of these leaking components; and identify whether any of these components have ever lost their intended function prior to repair of the leaking component.

NMC Response:

The NUREG-1801 use of the word "may" in describing daily inspection of leaking non-ASME bolted joints provides for lack of clarity regarding program requirements. There is no requirement to periodically inspect leaking pressure retaining joints which fall under the scope of ASME Section XI. Documentation of leaking components is done via the corrective action process or the corrective maintenance process. All corrective action and corrective maintenance requests which have the potential to affect equipment operability are reviewed by a Senior Reactor Operator. The condition is evaluated and the appropriate actions relative to the significance of the condition are taken. The appropriate actions may include an expeditious repair, scheduled future repair with periodic monitoring until the repair is completed, scheduled future repair with no periodic monitoring, or no specific actions. Significant increases in leakage would likely be noted via operator rounds during normal plant operations and/or observation of reduction of inventory in closed systems or increased flows in various drainage systems. Requiring formal daily inspections of leaking non-ASME bolted joints absent the evaluation and prioritization process afforded by the corrective maintenance and corrective action process and without ALARA considerations is inappropriate. As noted in Section B2.1.4 of the LRA, a review of plant-specific operating experience identified no instances of loss of intended function of a component or system due to fastener degradation.

NRC Question RAI B.2.1.4-5:

In the Acceptance Criteria of the Bolting Integrity Program, the applicant indicates that cracks in component support bolting will be repaired when scheduled as part of the plant maintenance and/or corrective action process. GALL Program XI.M18 indicates that cracked bolts in component supports should be replaced immediately. The staff requests the applicant to identify how the plant maintenance and/or corrective action process determines when cracked component support bolting is replaced; and identify whether any of these components have ever lost their intended function prior to repair of the cracked bolting.

NMC Response:

The corrective action process is used upon discovery of cracked bolting in a support that is in-scope in accordance with 10 CFR 54.4. As a result an immediate operability determination for cracked support bolting is made by a Senior Reactor Operator. As part of this operability determination an Operability Recommendation can be requested from Engineering to further document the support's functionality and operability. This operability determination process is used for any corrective action or corrective maintenance documented degradation which has the potential to affect equipment operability. The corrective action process will ensure the appropriate response including shutdown of operating units if necessary to comply with PBNP Technical Specifications, other CLB requirements or to establish the conditions necessary to allow the repair.

The degradation documented by the corrective action process for an operable support with a cracked bolt will be evaluated and appropriate priority set for repair or replacement. In all cases, the appropriate response for cracked support bolts is to initiate actions in accordance with PBNP's corrective action process which includes corrective maintenance.

As noted in Section B2.1.4 of the LRA, a review of plant-specific operating experience identified no instances of loss of intended function of a component or system due to fastener degradation.