NMC
Committed to Nuclear Excellence

Point Beach Nuclear Plant
Operated by Nuclear Management Company, LLC

June 9, 2005

NRC 2005-0044
10 CFR 54

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

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 March 30, 2005, the Nuclear Regulatory Commission (NRC) requested additional information regarding Aging Management of Reactor Coolant System (Section 3.1 of the LRA); Aging Management of Containments, Structures and Component Supports, (Section 3.5 of the LRA); and Aging Management Programs (Section B2.1 of the LRA). During an April 28, 2005, NMC telephone conference with the NRC staff, additional time was granted to respond to these questions in order to include additional clarifications in the response. 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.

Summary of Commitments

This letter contains the following new commitments:

1. NMC will use enhanced volumetric examination to detect and size cracks or a plant- or component-specific flaw tolerance evaluation to demonstrate that cast austenitic stainless steel (CASS) primary loop elbows potentially susceptible to thermal embrittlement have adequate fracture toughness.

2. NMC will age manage the steam generator (SG) feedrings, J-nozzles, and feedring supports using the Water Chemistry Control Program and the Steam Generator Integrity Program.

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3. The Structures Monitoring Program will examine below-grade concrete when it is exposed by excavation for signs of degradation from aggressive chemical attack or corrosion of embedded steel, during the period of extended operation.

   Periodic monitoring of ground water chemistry (pH, chlorides, sulfates) will continue to be performed during the period of extended operation to ensure the environment remains non-aggressive. The frequency of monitoring ground water chemistry (pH, chlorides, sulfates) will be at least once every 5 years.

This letter contains the following revisions to existing commitments (additions are bolded; deletions are strikethrough):

1. Draft SER, Appendix A, Page A-16, Commitment Number 72:

   If degradation is detected by the Flow Accelerated Corrosion Program such that the wall thickness is less than or equal to 87.5 % of nominal wall thickness for safety related piping~~, or 60 % of nominal wall thickness for non-safety related piping,~~ additional examinations will be performed in adjacent areas to bound the thinning. ~~The sample size will also be expanded when inspection results indicate that a component has a remaining service life less than one operating cycle. This covers situations where the code minimum allowable wall thickness may be greater than 60 % of nominal wall thickness.~~ **For both safety related and non-safety related piping, additional examinations will be performed in adjacent areas to bound the thinning if the remaining service life, based on the code minimum allowable wall thickness, is less than one operating cycle. The sample size will also be expanded for non-safety related piping if degradation is detected such that the wall thickness is less than or equal to 60% of nominal wall thickness. This covers situations where the code minimum allowable wall thickness may be less than 60% of nominal wall thickness for non-safety related piping.**

I declare under penalty of perjury that the forgoing is true and correct. Executed on June 9, 2005.

for 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
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.

Aging Management of Reactor Coolant System

<u>NRC Question RAI 3.1.1-1</u>:

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

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Of particular concern is the statement that:

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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- or component-specific flaw tolerance evaluation."

The staff requests the applicant to clarify if it 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 of 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.

NMC Response:

NMC intends to follow the recommendation of GALL Report XI.M12. NMC will use enhanced volumetric examination to detect and size cracks or a plant- or component-specific flaw tolerance evaluation to demonstrate that CASS primary loop elbows potentially susceptible to thermal embrittlement have adequate fracture toughness.

NRC Question 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.

NMC Response:

In the PBNP License Renewal Application dated February 25, 2004, NMC did not include the PBNP steam generator (SG) feedrings and associated J-nozzles for aging management. This was based on previous NRC-approved applicant positions (Westinghouse PWRs) that also did not include the feedring and J-nozzles for aging management. NUREG-1801 includes line item IV D1.3-a for the upper assembly, separators and feedwater inlet ring and support but states that this form of degradation has been detected only in certain Combustion Engineering System 80 SGs.

The NRC Information Notice (IN 91-19) on this issue stated that a licensee found several pieces of carbon steel debris during a routine inspection of the secondary side of the tubesheet of one steam generator. During further inspection of the internal components of this and the other SG, the licensee found material missing from the lower portion of the feedring at its intersection with the distribution box, surface cracks in the heat-affected zone at the toe of the weld at that intersection, erosion and corrosion indications on the interior surfaces of the distribution boxes, erosion of the vent assemblies, "T" section tops missing from the vent assemblies, and deformation of several U-bolt supports. The licensee determined the root cause contributing to the degradation of the feedwater distribution system piping to be inadequate design of the feedring and feedring supports. The design did not adequately consider the thermal stresses resulting from normal operating conditions, in particular the batch process of auxiliary feedwater addition during startup operations. In addition, the design of the vent assembly had not properly considered the potential for erosion and corrosion resulting from localized high velocity flow.

PBNP has Westinghouse Model 44F (Unit 1) and Model 47F (Unit 2) SGs. The feedring design in PBNP's SGs does not include a distribution box or vent assemblies. The PBNP J-nozzles are constructed of Alloy 600, which are not susceptible to flow accelerated corrosion. The feedrings are constructed of carbon steel, but plant-specific operating experience has shown that PBNP feedrings have not exhibited wall thinning. Previous SG secondary side inspections at PBNP have identified no feedring or feedring support damage. Therefore, NMC concluded that this line item was not applicable to PBNP.

None the less, the PBNP J-nozzles, feedrings, and feedring supports have been added to the scope of license renewal to address a potential non-safety affecting safety issue, where failures of these components could affect the safety-related SG tubing below. Therefore, the aging management programs for these components need to ensure that the feedring components stay in-place and not fall on the SG tubes.

NMC will age-manage the SG feedrings, J-nozzles, and feedring supports using the Water Chemistry Control Program and the Steam Generator Integrity Program. The Steam Generator Integrity Program provides for various inspections of the secondary side of the SGs, which will provide verification that aging effects are not progressing, thereby ensuring that the feedring and J-nozzles remain in-place.

Aging Management of Containments, Structures, and Component Supports

NRC Question 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 penetrations have had a fatigue review and are bounded by line item 3.5.1-01/II.A3.1-b. Therefore, the applicant did not address 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 line item 3.5.1-01/II.A3.1-b to manage cracking due to cyclic loading.

NMC Response:

NMC has interpreted NUREG-1801 Vol. 1, Table 5, Items 3.5.1-1 and 3.5.1-2 to be addressing a total of two aging effects: fatigue or cyclic loading AND stress corrosion cracking (SCC). The aging effect of *fatigue* (II.A3.1-b) is the same as *cyclic loading* (II.A3.1-c). The NUREG-1801 line item II.A3.1-b states that "(Only if CLB fatigue analysis exists)" versus II.A3.1-c, which states "(CLB fatigue analysis does not exist)." One concludes that this is an OR logic statement with respect to *fatigue/cyclic loading*. PBNP does have a current licensing basis (CLB) fatigue analysis; therefore, it credits NUREG-1801 Item II.A3.1-b and not II.A3.1-c with respect to *fatigue/cyclic loading*. The LRA directs the reader to the TLAA Section 4.3.11 for the fatigue analysis discussion.

SCC is the other aging effect that is addressed in LRA Table 3.5.1-2 and Item II.A3.1-d. LRA Section 3.5.2.2.1.7 goes on to say that SCC is not an applicable aging mechanism for these components at PBNP:

> "SCC is not an applicable aging mechanism for penetration sleeves, bellows and dissimilar metal welds. The carbon steel components within penetrations are not susceptible to SCC. The stainless steel components require both a high temperature (>140°F) and exposure to an aggressive chemical environment (e.g., exposure to chlorides). The bellows at Point Beach are not exposed to aggressive chemical environments."

The aging management of the containment penetration sleeves is covered by the ASME Section XI, Subsection IWE and IWL Inservice Inspection Program and the Boric Acid Corrosion Program, as presented in LRA Table 3.5.2-1, pages 3-437 and 3-438. All electrical and mechanical penetrations receive a VT-3 visual examination which looks for aging effects of any form including cracking of welds. The examinations are in accordance with IWE Category E-A, Item E1.12.

PBNP does not have any penetration bellows within the scope of License Renewal. The bellows associated with the containment penetrations are on the exterior of the containment and are not considered part of the pressure boundary. It should also be noted that IN 92-20 pertains to double ply bellows used with metal containments. Double ply bellows can be local leak rate tested by testing between the plies. The bellows at PBNP are single ply bellows.

NRC Question 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.

<u>NMC Response</u>:

(1) The following table compares the below-grade ground water sample testing results performed from 2000 to 2004 to the NUREG-1801 values indicating an aggressive environment.

Ground Water Environment Monitoring Data for PBNP

Parameter	PBNP Results	Aggressive Limit
pH	6.2 - 11.2	<5.5
Chlorides	5 - 100 ppm	>500 ppm
Sulfates	22 - 380 ppm	>1500 ppm

(2) Aging management of concrete is discussed in, among other places, LRA Section 2.1.1.3.2, Concrete Aging Management Program (ISG-03). The programs that manage the potential aging effects include ASME Section XI, Subsections IWE and IWL Inservice Inspection Program and the Structures Monitoring Program. The plant's concrete mix designs specified that the total air content shall not be less than 3% or more than 5% of concrete by volume to address the freeze-thaw aging effect. NUREG-1801 does not require further evaluation of the aging management programs if the stated conditions are satisfied or a non-aggressive environment exist. The concrete aging discussions in the LRA are cross referenced to the NUREG-1801 Volume II line item in the table below. Certain line items from the table pertain to concrete aging of containments and Class 1 structures.

LRA/NUREG-1801 Concrete Aging Cross Reference

Table 3.5.1 Line Item	Aging Effect/Mechanism	LRA Discussion Location	NUREG-1801, Vol. II
	Containment		
3.5.1-07	Leaching of Calcium Hydroxide, Aggressive Chemical Attack, Corrosion of Embedded Steel	3.5.2.2.1.1	II.A1.1-b, -c, & -e
3.5.1-08	Settlement	3.5.2.2.1.2	II.A1.1-f
3.5.1-09	Porous Concrete Subfoundation	3.5.2.2.1.2	II.A1.1-g
3.5.1-10	Elevated Temperature	3.5.2.2.1.3	II.A1.1-h
3.5.1-16	Freeze-thaw, Reaction with Aggregate	Table 3.5.1	II.A1.1-a & -d
	A1. Group 1 Structures (typical)		
3.5.1-20	Accessible Concrete – Various	3.5.2.2.2.1	III.A1.1-a, -b, -c, -d, -f, -h, & -j
3.5.1-21	Inaccessible Concrete – Various	3.5.2.2.2.2	III.A1.1-e & -g
3.5.1-22	CWPH – Abrasion, Cavitation	Table 3.5.1	III.A6.1-h

A detailed review of the Structures Monitoring Program (LRA Section B2.1.20) was performed during the March 2005 NRC Region III License Renewal Inspection. The results of this review are documented in NRC Inspection Report 2005-005 dated May 2, 2005. During a review of NRC Inspection Report 2005-005, it was noted that NMC needed to clarify its intent for the inspections of normally inaccessible concrete and the periodic monitoring of ground water chemistry. The Structures Monitoring Program will examine below-grade concrete when it is exposed by excavation for signs of degradation from aggressive chemical attack or corrosion of embedded steel, during the period of extended operation. Periodic monitoring of ground water chemistry (pH, chlorides, sulfates) will continue to be performed during the period of extended operation to ensure the environment remains non-aggressive. (Reference LRA Section 3.5.2.2.2.2 and the response to RAI 3.5-10 provided by letter from NMC to NRC dated August 26, 2004 (NRC 2004-0086)). The frequency of monitoring ground water chemistry (pH, chlorides, sulfates) will be at least once every 5 years.

Flow Accelerated Corrosion Program

NRC Question 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, on 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.

NMC Response:

During the March 2005 NRC Region III License Renewal Inspection, a detailed review of the Flow Accelerated Corrosion Program (LRA Section B2.1.11) was completed. As a result of that review and discussions between the NRC Region III Inspection Team, the NRC License Renewal Branch, and NRC Division of Engineering personnel, a clarification to LRA Section B2.1.11, "Flow-Accelerated Corrosion Program," was provided by letter from NMC to NRC dated April 8, 2005 (NRC 2005-0037). Based upon discussions with the NRC staff on May 3, 2005, a revision to the April 8, 2005, letter was identified as being needed to clarify the intent of the sample expansion criterion. The following text replaces the PBNP LRA Section B2.1.11, "Flow Accelerated Corrosion Program," discussion under the program element "Monitoring and Trending" (Page B-123) as provided in the April 8, 2005, letter (additions are double underlined; deletions are strikethrough):

Monitoring and Trending

CHECWORKS code is used to predict component degradation in systems susceptible to flow-accelerated corrosion (FAC). Plant data, including material composition, system flow characteristics, and operating conditions are also important in determining the remaining service life, which is recalculated after each inspection.

CHECWORKS is acceptable because it provides a bounding analysis for FAC. The inspection schedule developed on the basis of the results of this predictive code provides reasonable assurance that adequate wall thickness will be maintained between inspections.

If degradation is detected such that the wall thickness is less than or equal to 87.5% of nominal wall thickness for safety related piping, ~~or 60 % of nominal wall thickness for non-safety related piping,~~ additional examinations will be performed in adjacent areas to bound the thinning. ~~The sample size will also be expanded when inspection results indicate that a component has a remaining service life less than one operating cycle. This covers situations where the code minimum allowable wall thickness may be greater than 60 % of nominal wall thickness.~~ For both safety related and non-safety related piping, additional examinations will be performed in adjacent areas to bound the thinning if the remaining service life, based on the code minimum allowable wall thickness, is less than one operating cycle. The sample size will also be expanded for non-safety related piping if degradation is detected such that the wall thickness is less than or equal to 60% of nominal wall thickness. This covers situations where the code minimum allowable wall thickness may be less than 60% of nominal wall thickness for non-safety related piping. The expansion of the sample size should include a minimum of the next two most susceptible components in that CHECWORKS line, any component within two pipe 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.

This element includes exceptions to the corresponding NUREG-1801 aging management program element. NUREG-1801 states: "If degradation is detected such that the wall thickness is less than the minimum predicted thickness, additional examinations are performed in adjacent areas to bound the thinning." Literal interpretation of this sample expansion criteria is not practical in many cases. If very little degradation is predicted, measured wall thickness may be less than the predicted thickness even though the calculated life of the affected component may exceed the operating life of the plant. In this case, sample expansion would not be warranted.

The FAC program at PBNP implements the EPRI guidelines in NSAC-202L-R2, which recommends increasing the sample size when inspections of the sample detect significant FAC wear. In the PBNP FAC program, significant FAC wear is defined as FAC resulting in a wall thickness of less than or equal to 87.5% of nominal wall thickness for safety related piping. ~~or 60 % of nominal wall thickness for non-safety related piping. In addition, the sample size will be expanded when inspection results indicate that a component has a remaining service life less than one operating cycle. This covers situations where the code minimum allowable wall thickness may be greater than 60 % of nominal wall thickness.~~ For both safety related and non-safety related piping, additional examinations will be performed in adjacent areas to bound the thinning if the remaining service life, based on the code minimum allowable wall thickness, is less than one operating cycle. The sample size will also be expanded for non-safety related piping if degradation is detected such that the wall thickness is less than or equal to 60% of nominal wall thickness. This covers situations where the code minimum allowable wall thickness may be less than 60% of nominal wall thickness for non-safety related piping. This *criterion for sample expansion is acceptable because it* specifies a wall thickness criterion and requires projection of inspection results to the next inspection opportunity consistent with industry guidance. Therefore, PBNP meets the intent of this NUREG-1801 aging management program element.

This clarification to the text for the Flow-Accelerated Corrosion Program demonstrates reasonable assurance that structural integrity will be maintained between inspections.

Aging Management Programs

NRC Question RAI B2.1:

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

TABLE 1 - Point Beach Nuclear Plant LRA Relief Requests	
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

NMC Response:

During a telephone conference held between NRC staff and NMC personnel on May 26, 2005, the Staff granted additional time to respond to this question.