January 19, 2005

LICENSEE: Nuclear Management Company, LLC

FACILITY: Point Beach Nuclear Plant, Units 1 and 2

SUBJECT: SUMMARY OF TELEPHONE CONFERENCE HELD ON JANUARY 10, 2005, BETWEEN THE U.S. NUCLEAR REGULATORY COMMISSION AND NUCLEAR MANAGEMENT COMPANY, LLC, CONCERNING REQUESTS FOR ADDITIONAL INFORMATION PERTAINING TO THE POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2, LICENSE RENEWAL APPLICATION

The U.S. Nuclear Regulatory Commission staff (the staff) and representatives of Nuclear Management Company, LLC (NMC) held a telephone conference on January 10, 2005, to discuss and clarify the staff's requests for additional information (RAIs) concerning the Point Beach Nuclear Plant, Units 1 and 2, license renewal application. The conference call was useful in clarifying the intent of the staff's RAIs.

Enclosure 1 provides a listing of the meeting participants. Enclosure 2 contains a listing of the RAIs discussed with the applicant, including a brief description on the status of the items. Enclosure 3 contains draft responses provided by the applicant.

The applicant had an opportunity to comment on this summary.

/RA/
Verónica M. Rodríguez, 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

Enclosures: As stated

cc w/encls: See next page

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Enclosure 1 provides a listing of the meeting participants. Enclosure 2 contains a listing of the RAIs discussed with the applicant, including a brief description on the status of the items. Enclosure 3 contains draft responses provided by the applicant.

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DISTRIBUTION: Note to Licensee: NMC, LLC, Pt. Beach Nuclear Plant, Units 1 and 2, Re: Summary of Telephone Conference Held on January 10, 2005

ML050260237

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LIST OF PARTICIPANTS FOR TELEPHONE CONFERENCE
TO DISCUSS THE POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2
LICENSE RENEWAL APPLICATION

JANUARY 10, 2005

Participants	Affiliations
J. Knorr	Nuclear Management Company, LLC
M. Morgan	Nuclear Regulatory Commission
G. Suber	Nuclear Regulatory Commission
V. Rodriguez	Nuclear Regulatory Commission

Enclosure 1

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

January 10, 2005

The U.S. Nuclear Regulatory Commission staff (the staff) and representatives of Nuclear Management Company, LLC (NMC) held a telephone conference call on January 10, 2005, to discuss and clarify the staff's requests for additional information (RAIs) concerning the Point Beach Nuclear Plant, Units 1 and 2, license renewal application (LRA). The following RAIs were discussed during the telephone conference call.

4.4 Fracture Mechanics Analysis

RAI 4.4.2 (Reactor Coolant Pump Flywheel Analysis)

The staff needs explanation as to the number of reactor coolant pump (RCP) start/stop cycles that are assumed in the 60-year RCP flywheel fatigue crack growth assessment for the Point Beach Units.

Discussion: The applicant clarified their draft response. The applicant will provide their formal response in writing.

B 2.0 Aging Management Programs

RAI B 2.1.6-1 (Boric Acid Corrosion Program)

The staff seeks additional clarification regarding the list of components that are within the scope of the Boric Acid Corrosion Program and the process the applicant uses to augment the list of components within the scope of the AMP based on pertinent industry experience. This is RAI B2.1.6-1. Specifically, the staff requested the following actions of the applicant:

- Submittal of a discussion on how NMC's responses to NRC Bulletin 2002-01, dated March 29, 2002, and May 16, 2002; responses to the NRC's RAIs on the bulletin, dated January 17, 2003; response to NRC Bulletin 2003-02, dated September 19, 2003; responses to NRC Order EA-03-009, dated March 3, 2003, April 11, 2003, and April 18, 2003; and response to NRC Bulletin 2004-01, dated May 28, 2004 have been used to update the list of component locations and types of visual inspections credited within the scope of the Boric Acid Corrosion Program or within the scope of other aging management programs (AMPs) that provide for implementation of similar or more conservative types of inspections.

- If the responses were used to supplement the scope of the Boric Acid Corrosion Program or other AMPs, identification of the component locations that have been added to the scope of the program and clarfication of the type of visual examinations (i.e., specification on whether VT-1, VT-2 or VT-3 will be used and whether the visual examinations will be enhanced, bare-surface, qualified, etc.) that will be implemented on those components within the current scope of the program.

Discussion: The applicant clarified their draft response. The applicant will provide their formal response in writing.

RAI B 2.1.18-1 (Reactor Vessel Surveillance Capsule Program)

GALL Program XI.M31 suggested that standby capsules are to be removed and placed in storage. Even though the capsules do not contain limiting material, these standby capsules provide general embrittlement trends and provide assurance that current embrittlement methodologies apply to Point Beach. Leaving the capsules in the vessel, further exposure would not provide any meaningful data. Please justify your decision of not removing the capsule and keep it in storage.

Discussion: The applicant clarified their draft response. The applicant will provide their formal response in writing.

RAI B 2.1.23-1 (Thimble Tube Inspection Program)

Section B2.1.23 indicates that eddy current examinations are performed on a periodicity consistent with the severity of wear damage for each thimble tube. The frequency of inspections is based on the maximum wall loss noted in a region of active wear and the projected wear which would occur based on a known wear rate.

1. Identify the wear rate that is currently being used and how did you calculate the wear rate. Based on this wear rate, how were the inspection intervals determined to ensure that wear resulting from flow induced vibration does not result in the wall thickness below the minimum required thimble tube integrity?

2. Specify the NDE uncertainty that is used in the calculations along with a justification for the NDE uncertainty value assumed in the calculation. Note that we would like the NDE uncertainty to be specified as a given percentage of the nominal wall thickness for the thimble tubes.

The applicant's Operating Experience identified certain problems related to inspection deferrals, calculation methodology, and record retention.

Explain the problems in detail and how and when are you going to address the issues.

Discussion: The applicant clarified their draft response. The applicant will provide their formal response in writing.

ENCLOSURE

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

NRC Question RAI 4.4.2 (Reactor Coolant Pump Flywheel Analysis):

The staff needs explanation as to the number of reactor coolant pump (RCP) start/stop cycles that are assumed in the 60-year RCP flywheel fatigue crack growth assessment for the Point Beach Units.

NMC Response:

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

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

The analysis assumed 6000 cycles of RCP starts and stops for a 60-year plant operating life. The assumption of 6000 cycles is extremely conservative for PBNP. The following is an estimate of RCP starts and stops cycles for PBNP based on discussions with Operations personnel:

1st 10-years of operation

10 * 5-start/stop cycles = 50 cycles (fill and vent)

2nd 10-years of operation

10 * 5-start/stop cycles = 50 cycles (fill and vent)
10 *10-start/stop cycles = 100 cycles (SG crevice flushing)

Remaining 40 years of operation

40 * 5-start/stop cycles = 200 cycles (fill and vent)

The total estimated normal RCP start / stop cycles is approximately 400 cycles for a 60-year operating period. Assume an additional 100 start / stop cycles occur to account for maintenance activities and inadvertent pump trips. The total number of pump start / stop cycles then becomes approximately 500 cycles. Even using very conservative start / stop cycle estimates (a refueling every year), the total number of cycles is less than 10% of the assumed number of cycles in the WCAP-14535-A analysis.

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

NRC Question RAI 2.1.6.-1 (Boric Acid Corrosion Program):

The staff seeks additional clarification regarding the list of components that are within the scope of the Boric Acid Corrosion Program and the process the applicant uses to augment the list of components within the scope of the AMP based on pertinent industry experience. This is RAI B2.1.6-1. Specifically, the staff requested the following actions of the applicant:

Submittal of a discussion on how NMC's responses to NRC Bulletin 2002-01, dated March 29, 2002, and May 16, 2002; responses to the NRC's RAIs on the bulletin, dated January 17, 2003; response to NRC Bulletin 2003-02, dated September 19, 2003; responses to NRC Order EA-03-009, dated March 3, 2003, April 11, 2003, and April 18, 2003; and response to NRC Bulletin 2004-01, dated May 28, 2004 have been used to update the list of component locations and types of visual inspections credited within the scope of the Boric Acid Corrosion Program or within the scope of other aging management programs (AMPs) that provide for implementation of similar or more conservative types of inspections.

If the responses were used to supplement the scope of the Boric Acid Corrosion Program or other AMPs, identification of the component locations that have been added to the scope of the program and clarification of the type of visual examinations (i.e., specification on whether VT-1, VT-2 or VT-3 will be used and whether the visual examinations will be enhanced, bare-surface, qualified, etc.) that will be implemented on those components within the current scope of the program.

NMC Response:

PBNP's response to NRC Bulletin 2004-01 stated that no Alloy 82/182/600 materials exist in the Unit 1 and Unit 2 pressurizers and the response did not require any augmentation of the Boric Acid Corrosion AMP. PBNP's response to NRC Bulletin 2002-01 concluded that the existing boric acid program was in compliance with Generic Letter 88-05 (LRA Page B-74). Enhancements to the Reactor Coolant System Alloy 600 Inspection AMP will include the development of new implementing documents to meet the commitments made in response to NRC Bulletin 2003-02 and revised NRC Order EA-03-009 (LRA Page B-164).

In response to NRC Bulletin 2003-02, a bare metal visual examination of the RPV lower head and BMI nozzles was performed by VT-2 qualified individuals during the Unit 2 October 2003 outage and the Unit 1 April 2004 refueling outage. The examinations were performed directly with a resolution capability of VT-1 and had acceptable results. The insulation on the RPV lower head has been modified and now provides access ports. A bare metal visual examination is now performed by VT-2 qualified individuals via the access ports each refueling outage. The visual examinations will continue until industry experience, changes to the ASME code, or a change in regulatory requirements justify a change to the inspection frequency or method. Additionally, a VT-2 examination subject to the pressure and temperature requirements, and other requirements specific to the performance of an ASME Section XI code-required system pressure test is conducted.

The First Revised Order EA-03-009, dated February 20, 2004, requires bare metal visual examination of the RPV head surface and either UT of the head penetration nozzle, EC or dye penetrant testing of the wetted surface of each J-groove weld and RPV nozzle base, or combinations of these non visual NDE methods. The bare metal visual examination is performed by VT-2 qualified individuals when the reactor vessel head is removed and located in the lay down area. The revised Order also requires visual inspections each refueling to identify potential boric acid leaks from pressure retaining components above the RPV head. These inspections are satisfied by the performance of a VT-2 inspection of the CRDM mechanism, conoseals, piping etc. located above the RPV head subject to the pressure and temperature requirements, and other requirements specific to the performance of an ASME Section XI code-required system pressure test.

PBNP is in compliance with the revised NRC Order for PBNP-2 and is in compliance with the revised Order for PBNP-1 except for the relaxation regarding the examination distance below the J-groove weld on 17 RPV nozzles that was approved by the NRC in an SER dated June 4, 2004. The relaxation for PBNP-1 will remain in effect until the end of the current operating cycle, at which time the RPV head will be replaced.
Note: The Unit 1 and Unit 2 RPV heads are scheduled for replacement in 2005.

NRC Question RAI 2.1.18-1 (Reactor Vessel Surveillance Capsule Program):

GALL Program XI.M31 suggested that standby capsules are to be removed and placed in storage. Even though the capsules do not contain limiting material, these standby capsules provide general embrittlement trends and provide assurance that current embrittlement methodologies apply to Point Beach. Leaving the capsules in the vessel, further exposure would not provide any meaningful data. Please justify your decision of not removing the capsule and keep it in storage.

NMC Response:

While there are no current plans to remove the standby capsules from the reactor vessel, the standby capsules might be removed at sometime in the future to support industry needs. The lead factor for these locations is low enough to allow extended neutron exposure. Removal of these capsules will not directly support demonstration of adequate Upper Shelf Energy (USE) and Pressurized Thermal Shock (PTS) margins for the PBNP-1 and PBNP-2 reactor vessels, and consequently removal was not required by the AMP. The Reactor Vessel Surveillance Aging Management Program will require that all withdrawn surveillance capsules not discarded as of August 31, 2000 be placed in storage for potential future use.

NRC Question RAI 2.1.23-1 (Thimble Tube Inspection Program):

Section B2.1.23 indicates that eddy current examinations are performed on a periodicity consistent with the severity of wear damage for each thimble tube. The frequency of inspections is based on the maximum wall loss noted in a region of active wear and the projected wear which would occur based on a known wear rate.

Identify the wear rate that is currently being used and how did you calculate the wear rate. Based on this wear rate, how were the inspection intervals determined to ensure that wear resulting from flow induced vibration does not result in the wall thickness below the minimum required thimble tube integrity?

Specify the NDE uncertainty that is used in the calculations along with a justification for the NDE uncertainty value assumed in the calculation. Note that we would like the NDE uncertainty to be specified as a given percentage of the nominal wall thickness for the thimble tubes.

The applicant's Operating Experience identified certain problems related to inspection deferrals, calculation methodology, and record retention.

Explain the problems in detail and how and when are you going to address the issues.

NMC Response:

Based on a conservative calculation of the collapse strength for a thimble tube, an incore thimble tube is assumed to collapse at design pressure after it has worn 83% through wall. This calculation assumes uniform wear around the circumference of the tube with no credit taken for any reinforcement around the worn area. For a typical tube, fretting wear will be localized resulting in a large volume of sound tube material surrounding the worn area. Consequently, the wall loss would have to be in excess of 83% of nominal wall thickness before collapse would occur.

To ensure that tube failure does not occur between inspections, tubes exhibiting a wall loss in excess of 60% will be capped and taken out of surface. Tubes will also be capped if their indicated wall loss plus the predicted wall loss, between inspections, exceed 60%. The 60% capping limit was established as follows:

FACTOR	% Wall Loss
Maximum allowable wall loss	83%
Error in eddy current inspection	-10%
Uncertainty in wall loss geometry	-10%
Capping Limit	63%

The error in the eddy current inspection results is accounted for by reducing the maximum allowable wall loss by 10%, which is consistent with the accuracy assumed in the industry for these types of eddy current inspections (i.e., industry standard). The maximum allowable wall loss is further reduced by an additional 10% to account for any uncertainty introduced by wall loss geometry. The capping limit is also established at 60% versus 63% to provide additional conservatism. Taken together, this provides an overall conservative approach for establishing the capping limit and calculating the inspection frequency.

To prolong the life of thimble tubes experiencing wear, a tube may be repositioned to move the worn area away from the lower core plate. This will move the degraded portion of the tube into an area where fretting is not occurring and place intact tube material in the area where fretting is occurring. This will extend the life of the thimble tube without undue risk of failure. The repositioning of a thimble tube is evaluated on a case-by-case basis. However, the previously discussed capping limit is adhered to in all cases.

Damage to thimble tubes has generally been observed as a decelerating phenomenon, where the progression of damage decreases with time. Therefore, a conservative method for calculating the wear rate would be to take the total wall loss over the life of the thimble tube divided by the operating time of the tube. Unless the wear has been found to increase during an inspection period, the assumption of wear decrease with time is the method used for determining the wear rate. The wear rate is determined for each thimble tube and wear location (i.e., thimble tubes may have fretting wear at multiple locations).

The frequency of inspection is based on the maximum wall loss (i.e., worst case flaw depth) noted in a region of active fretting and the projected wear that would occur based on a known wear rate. The maximum inspection frequency for thimble tubes is determined as follows:

$$F = \frac{WL_{max} - WL_{meas}}{WR}$$

WL_{max} = Capping Limit (%)

WL_{meas}= Wall Loss Measured (%)

WR = Wear Rate (%/year)

Inspections are normally performed during the outage prior to the outage that would be identified as the one before the lowest minimum thimble tube life calculation or at least every 6 years, to ensure conservative testing intervals. For example: If the lowest minimum thimble tube life is 5.5 years, the next inspection would be at 3 years, one outage prior to the 4.5 year outage. If the lowest minimum thimble tube life is 12.8 years, the next inspection would be at 6 years. However, this does not preclude thimble tube inspections being scheduled more frequently to ensure conservatism.

The PBNP Unit 1 thimble tubes were last inspected during the 2001 refueling outage with the lowest minimum thimble tube life calculated at 10.85 years using a calculated wear rate of 4.33 %/year. The next inspection is scheduled at 4.5 years during the 2005 refueling outage, to ensure conservatism. Table 1 below provides the thimble tube inspection history and plan for PBNP Unit 1.

The PBNP Unit 2 thimble tubes were last inspected during the 2000 refueling outage with the lowest minimum thimble tube life calculated at 10.71 years using a calculated wear rate of 2.33 %/year. The next inspection is scheduled at 4.5 years during the 2005 refueling outage, to ensure conservatism. Table 2 below provides the thimble tube inspection history and plan for PBNP Unit 2.

PBNP identified several concerns related to inspection deferrals, calculation methodology, and record retention. As noted in Table 1, the inspection of the PBNP Unit 1 thimble tubes was deferred from the 1999 refueling outage to the 2001 refueling outage, resulting in an inspection interval of 5 years. As noted in Table 2, the inspection of the PBNP Unit 2 thimble tubes was also deferred from the 1996 refueling outage to the 2000 refueling outage, resulting in an inspection interval of 5 years. However, in no case did the inspection interval exceed the lowest minimum thimble tube life on either unit. The concern identified the fact that these inspections were deferred without a documented evaluation showing that there would be no problems with the extended inspection interval. The program implementing document currently requires an engineering evaluation or calculation to be performed demonstrating, with a high degree of confidence, that there would be no thimble tube problems with an extended inspection interval prior to a deferral being allowed.

Following the PBNP Unit 2 thimble tube inspection performed during the 2000 refueling outage, it was discovered that the eddy current contractor had reported the results using an averaged flaw depth method (i.e., mixed frequency amplitude wall loss) instead of the worst case flaw depth method (i.e., maximum wall loss) described above. The results were subsequently reanalyzed using the worst case flaw depth method, which resulted in the lowest minimum thimble tube life being calculated at 10.71 years. While resolving this issue, it was also discovered that the worst case flaw depth method has been largely replaced in the industry with the mixed frequency amplitude wall loss method. Therefore, future inspections may be analyzed using both of these methods to determine if PBNP should change to the current industry standard method.

The concern expressed regarding record retention involved copies of the official records being maintained by the Thimble Tube Condition Assessment Program Engineer for ease of reference. The previous Thimble Tube Condition Assessment Program Engineer had disposed of these working copies, since they were redundant to the official records being maintained. This increased the burden associated with locating the records whenever they were needed. The official records are retained and stored appropriately.

Therefore, the concerns related to inspection deferrals, calculation methodology, and record retention have been appropriately addressed.