June 6, 2005

Dennis Koehl
Site Vice President
Point Beach Nuclear Plant
Nuclear Management Company, LLC
6610 Nuclear Road
Two Rivers, WI 54241-9516

SUBJECT: POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2, ISSUANCE OF AMENDMENTS RE: LEAK-BEFORE-BREAK EVALUATION FOR PRIMARY LOOP PIPING (TAC NOS. MC1279 AND MC1280)

Dear Mr. Koehl:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 219 to Facility Operating License No. DPR-24 and Amendment No. 224 to Facility Operating License No. DPR-27 for the Point Beach Nuclear Plant (PBNP), Units 1 and 2, respectively. The amendments revise the PBNP, Units 1 and 2, Updated Final Safety Analysis Report to reflect the NRC staff's approval of the WCAP-14439-P, Revision 2 analysis entitled, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Point Beach Nuclear Plant, Units 1 and 2 for the Power Uprate and License Renewal Program," in response to your application dated November 5, 2003, as supplemented by letter dated April 22, 2004.

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Harold K. Chernoff, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-266 and 50-301

Enclosures:
1. Amendment No. 219 to DPR-24
2. Amendment No. 224 to DPR-27
3. Safety Evaluation

cc w/encls: See next page

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NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-266

POINT BEACH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 219
License No. DPR-24

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Nuclear Management Company, LLC (the licensee), dated November 5, 2003, as supplemented by letter dated April 22, 2004, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, by Amendment No. 219, Facility Operating License No. DPR-24 is hereby amended by changes to the Updated Final Safety Analysis Report (UFSAR). These changes reflect the NRC staff's approval of the WCAP-14439-P, Revision 2 analysis entitled, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Point Beach Nuclear Plant, Units 1 and 2 for the Power Uprate and License Renewal Program."

3. This license amendment is effective as of its date of issuance. Implementation of the amendment is the incorporation into the next UFSAR update, made in accordance with 10 CFR 50.71(e), of the changes to the description of the facility as described in the Point Beach Nuclear Plant, Units 1 and 2 application dated November 5, 2003, as supplemented by letter dated April 22, 2004, and as evaluated in the NRC staff's Safety Evaluation dated June 6, 2005.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

L. Raghavan, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the
Technical Specifications

Date of issuance: June 6, 2005

NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-301

POINT BEACH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 224
License No. DPR-27

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Nuclear Management Company, LLC (the licensee), dated November 5, 2003, as supplemented by letter dated April 22, 2004, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, by Amendment No. 224, Facility Operating License No. DPR-27 is hereby amended by changes to the Updated Final Safety Analysis Report (UFSAR). These changes reflect the NRC staff's approval of the WCAP-14439-P, Revision 2 analysis entitled, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Point Beach Nuclear Plant, Units 1 and 2 for the Power Uprate and License Renewal Program."

3. This license amendment is effective as of its date of issuance. Implementation of the amendment is the incorporation into the next UFSAR update, made in accordance with 10 CFR 50.71(e), of the changes to the description of the facility as described in the Point Beach Nuclear Plant, Units 1 and 2 application dated November 5, 2003, as supplemented by letter dated April 22, 2004, and as evaluated in the NRC staff's Safety Evaluation dated June 6, 2005

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

L. Raghavan, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the
Technical Specifications

Date of issuance: June 6, 2005

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 219 TO FACILITY OPERATING LICENSE NO. DPR-24

AND AMENDMENT NO. 224 TO FACILITY OPERATING LICENSE NO. DPR-27

NUCLEAR MANAGEMENT COMPANY, LLC

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-266 AND 50-301

1.0 INTRODUCTION

By letter to the U.S. Nuclear Regulatory Commission (NRC, Commission) dated November 5, 2003 (ADAMS ML033210549), as supplemented by letter dated April 22, 2004 (ADAMS ML041250404), Nuclear Management Company, LLC (the licensee), requested NRC review and approval of their analyses of the leak-before-break (LBB) evaluation for the primary loop piping for the Point Beach Nuclear Plant (PBNP), Units 1 and 2. The supplement dated April 22, 2004, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the Federal Register on February 7, 2005 (70 FR 6466). The licensee submitted an analysis entitled, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Point Beach Nuclear Plant, Units 1 and 2, for the Power Uprate and License Renewal Program," WCAP-14439-P, Revision 2. The licensee's analysis was submitted for NRC staff review and approval per the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, General Design Criteria (GDC) 4, "Environmental and dynamic effects design basis," which states, in part:

> However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

LBB evaluations developed using the analysis methodology contained in NUREG-1061, Volume 3, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Potential for Pipe Breaks," have been previously approved by the Commission as demonstration of an extremely low probability of piping system rupture.

2.0 REGULATORY EVALUATION

The requirement that nuclear power plant licensees consider the dynamic effects that could result from the rupture of sections of high energy piping (fluid systems that during normal plant operations are at a maximum operating temperature in excess of 200 EF and/or a maximum operating pressure in excess of 275 psig) is found in 10 CFR Part 50, Appendix A, GDC 4, which states, in part:

> Structures, systems, and components important to safety ... shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

As noted in Section 1.0 above, 10 CFR Part 50, Appendix A, GDC 4 permits the dynamic effects of some high energy piping ruptures to be excluded from facility design bases based upon the demonstration of an extremely low probability of piping system rupture. Consistent with GDC 4, the NRC staff accepted the LBB analysis methodology as an acceptable means by which this extremely low probability of piping system rupture could be demonstrated. The philosophy of LBB behavior for high energy piping systems was developed by the NRC in the early 1980s, used in certain evaluations stemming from Unresolved Safety Issue A-2, "Asymmetric Blowdown Loads on PWR [pressurized water reactor] Primary Systems," and then subsequently expanded for application toward resolving issues regarding defined dynamic effects from high energy piping system ruptures.

3.0 TECHNICAL EVALUATION

3.1 Summary of Identification of Analyzed Piping and Piping Material Properties

The following discussion contains information supplied by the licensee in its November 5, 2003, submittal. Included in the submittal was the report prepared by Westinghouse for the licensee WCAP-14439-P, Revision 2, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Point Beach Nuclear Plant, Units 1 and 2, for the Power Uprate and License Renewal Program." The licensee's evaluations demonstrate that dynamic effects of reactor coolant system (RCS) primary loop breaks need not be considered in the structural design basis of PBNP, Units 1 and 2.

The licensee's submittal identified and analyzed the RCS piping for LBB behavior verification. The primary loop piping is wrought stainless steel (SS) conforming to American Society for Testing and Materials (ASTM) A376 Type 316. The elbow fittings are cast stainless conforming to ASTM A351 CF8M. The as-built outside diameter and minimum wall thickness of the pipe are 34.21 inches and 2.50 inches, respectively.

For the material properties used in the primary loop LBB evaluations, Westinghouse used minimum and average room temperature tensile properties based on Certified Materials Test Report (CMTR) data. The minimum and average tensile properties at temperatures of interest (i.e., 541.4 EF and 605 EF) were calculated using the ratio of the 1989 American Society for Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (Code), Section III, properties at room temperature to the ASME Code properties at the temperatures of interest to scale the CMTR-based data. The modulus of elasticity variation with temperature was established based on ASME Code, Section III values. The minimum tensile properties were used by Westinghouse in the LBB critical flaw size determination, while the average tensile properties were used in the LBB leakage flaw size determination.

The preservice fracture toughness of CF8M cast SS in terms of J_{IC} (critical value of fracture toughness when the extension or growth of an existing flaw occurs) has been found to be very high but the material shows increased susceptibility to thermal aging at reactor operating temperature resulting in loss of fracture toughness. Westinghouse used the end-of-life fracture toughness value for the analysis based on the procedure developed by Argonne National Laboratory (ANL) under NRC-sponsored research. The ANL procedures produce conservative estimates of fracture toughness values 30 to 50 percent less than actual measured values. Westinghouse determined three critical heats of material in hot leg, cross-over leg and cold leg of the primary loop based on the fracture toughness values. The results from the ANL program indicate that the lower-bound fracture toughness of thermally aged cast SS is similar to that of submerged arc welds (SAWs). The applied value of the J-integral for a flaw in the weld regions will be lower than that in the base metal since the yield stress for the weld materials is much higher at that temperature. Therefore, weld regions are less limiting than the cast base metal.

3.2 Summary of General Aspects of the Licensee's LBB Analyses

The analyses provided by the licensee sought to address the following four principal areas that were consistent with the criteria established for LBB analysis acceptability in NUREG-1061, Volume 3: (1) demonstrate that the subject piping is a candidate for LBB analysis by showing that the piping is not particularly susceptible to active degradation mechanisms or atypical loading events; (2) establish the critical through-wall flaw size under which analyzed locations would be expected to fail under normal operation (NOP) plus safe-shutdown earthquake (SSE) loading conditions; (3) establish the leakage behavior of smaller through-wall flaws under NOP loads alone for each location; (4) evaluate the margin between the critical through-wall flaw size and an appropriate leakage through-wall flaw size and the stability of the through-wall leakage flaw.

3.3 Summary of the Licensee's Evaluation of Primary Loop Piping

The analysis of the primary loop piping that was submitted to the NRC staff as an attachment to the licensee's November 5, 2003, letter was prepared for the licensee by Westinghouse as reports WCAP-14439-P, Revision 2 and WCAP-14439-NP, Revision 0, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Point Beach Nuclear Plant, Units 1 and 2 for the Power Uprate and License Renewal Program." This section summarizes the Westinghouse results for the four subject areas noted in Section 3.2 above.

Initially, the licensee's submittal addressed the issue of potential piping degradation mechanisms and atypical loading conditions. Per the discussion of the limitations of LBB analyses in NUREG-1061, Volume 3, the LBB approach should not be considered when operating experience has indicated particular susceptibility to failure from the effects of corrosion, water hammer, or fatigue. Such mechanisms could cause the development of complex or extensive flaws in piping which significantly degrade its load carrying capacity while not propagating through-wall over a sufficient length to be detectable, or provide loads which are difficult to bound analytically. The licensee's submittal concluded that the primary loop piping like that at PBNP, Units 1 and 2, has not been shown to be particularly susceptible to the effects of water hammer, intergranular stress corrosion cracking (IGSCC), or erosion-corrosion.

Regarding the potential for fatigue cracking from mechanical and thermal loadings, an assessment of low cycle fatigue loading was carried out in the form of fatigue crack growth analysis at a bounding location for 60-year operation. The calculated fatigue crack growth for semi-elliptic flaws of circumferential orientation are very small regardless of material. High-cycle fatigue loads, primarily from pump vibrations, are managed through monitoring of the reactor coolant pump shaft vibration limits; inservice measurements have shown that the magnitude of the stresses associated with these vibrations are very low and below the level at which they would cause a significant concern.

An additional concern for primary loop piping was identified with the use of Inconel 82/182 weld metal between a SS safe-end and the steam generator (SG) nozzle. This material is known to be susceptible to primary water stress corrosion cracking (PWSCC). However, the inside bore of the nozzle buttering and the Inconel 82/182 weld is clad using Inconel 52/152 weld metal which is resistant to PWSCC. Since Inconel 82/182 is not exposed to primary coolant, PWSCC should not be a concern. Therefore, the primary loop piping at PBNP, Units 1 and 2 qualifies as a candidate for LBB analysis in accordance with NUREG-1061, Volume 3.

Next, the Westinghouse analysis evaluated the primary loop piping by developing the applied stresses under NOP and NOP plus SSE loading and determined the leakage and critical through-wall flaw size for various locations along the piping. In the determination of the applied stresses, the analysis generally included the tensile and bending stresses resulting from the internal pressure, deadweight, and thermal expansion, with SSE loads included when determining the loads associated with the critical flaw size evaluation.

In accordance with draft Standard Review Plan (SRP) 3.6.3 for the load combination, the deadweight, thermal expansion and/or thermal stratification, pressure, and SSE stresses were summed absolutely for the critical flaw size determination. The deadweight, thermal expansion, and pressure stresses were summed algebraically for the leakage flaw size determination.

For the purposes of LBB analyses, the critical flaw size can be defined as the longest preexisting through-wall flaw that could exist without growing unstably to double-ended pipe rupture under NOP plus SSE stresses. The LBB evaluation margins are to be demonstrated for the limiting locations. Candidate locations are designated as load critical with the highest stress or toughness critical with low toughness. For the welds on load critical SS piping and the toughness critical locations, a limit load analysis was employed. “Z” factor corrections to account for the generally lower toughness and lower load carrying capacity of SAWs were applied. The welds in this piping were assumed conservatively as SAWs to give highest “Z” factor to increase the applied load. This approach effectively predicts piping failure based on net section collapse of the cross-section that has been reduced by the through-wall cracked section. For the cast SS elbow fittings an evaluation was performed at the toughness critical locations to show that unstable crack extension will not occur for flaws two times as long as the leakage flaws.

The leakage flaw size for an LBB analysis is defined as the flaw size which, under NOP conditions, would leak 10 times the amount of fluid detectable by the facility’s leakage detection system. PBNP meets the criterion specified in Generic Letter 84-04, “Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops,” of being capable of detecting RCS leakage of 1 gallon per minute (gpm) within four hours. The factor of 10 is established in the LBB guidance of NUREG-1061, Volume 3, as the safety factor on leakage to account for uncertainties in calculating leakage from a through-wall crack. Leakage flaw sizes to yield a leak rate of 10 gpm were calculated at the governing locations.

3.4 NRC Staff Evaluation

The NRC staff reviewed the scope of the licensee’s LBB evaluation for the primary loop piping at PBNP, Units 1 and 2 on a plant-specific basis for the extended term of operation for a 60-year plant life and for the power uprate of up to 10.4 percent reactor power.

The LBB analysis consists of a leakage flaw size calculation using loading associated with normal operating conditions and a critical flaw size calculation using loading associated with faulted conditions. The pipe loading associated with normal operating conditions are axial forces and moments due to pressure, dead weight, and thermal expansion; the pipe loading associated with faulted conditions are axial forces and moments of normal operating conditions in conjunction with SSE and seismic anchor motion loads. In the licensee's critical flaw size calculation, the absolute sum method was used to add the individual axial forces and moments into the combined axial forces and moments. Therefore, the recommended margin on loads of 1.0 is satisfied as per SRP 3.6.3.

Based on the material property, operating condition, and loading information noted in the foregoing discussion, the licensee implemented its LBB evaluation. This process first required determination of the leakage flaw size (i.e., the length of a through-wall circumferential flaw at the critical locations in the analyzed piping segments that would generate a leakage rate of 10 gpm; 10 times 1 gpm, the amount of fluid detectable by the leakage detection system at PBNP, Units 1 and 2). Therefore, a margin of 10 exists between the calculated leak rate from the leakage flaw and the leak detection capability of 1 gpm. The licensee then determined the critical flaw sizes for the critical locations that would be predicted to lead to piping failure under the faulted loading conditions. These critical flaw size calculations were performed using plots of limit moment versus crack lengths. The critical flaw size corresponds to the intersection of this curve and the maximum load line. The last step in the licensee's evaluation process was the calculation of ratios (margins) between the critical flaw size and the leakage flaw size for the critical locations. The relationship between the critical flaw size and leakage flaw size results from the guidance in draft SRP 3.6.3 and NUREG-1061, Volume 3, which specifies that a margin of two should be maintained for an acceptable LBB evaluation. In WCAP-14439-P, Revision 2, for the hot leg, the crossover leg, and the cold leg (locations 2, 8, and 11 respectively), Westinghouse identified the limiting location with critical flaw sizes of 15.42, 15.98, and 13.75 inches and a leakage flaw size of 7.61, 7.91, and 6.88 inches, respectively, therefore, a margin of 2 based on J-integral evaluation for the standard LBB evaluation. Thus, the margin on flaw size is acceptable. Based on the aforementioned discussion, all of the LBB recommended margins are satisfied. The NRC staff has determined that a postulated through-wall crack in the primary loop piping would remain stable and would not cause a gross failure of the component.

In regard to the limitations of LBB analyses in NUREG-1061, Volume 3, the LBB approach should not be considered when operating experience has indicated particular susceptibility to failure from the effects of corrosion, water hammer, or fatigue. The NRC staff agrees with the licensee's evaluation that water hammer should not occur in the subject piping because of system design, testing, and operational considerations. The licensee's evaluation of the effects of low and high cycle fatigue were acceptable. The licensee addressed concerns regarding the impact of degradation mechanisms on the use of LBB for the primary loop piping. In the primary loop piping, all of the nozzle welds with the exception of the SG nozzle-to-safe end weld, are austenitic SS, and this material has not been shown to be susceptible to PWSCC. Further, there have been no occurrences of IGSCC for PWR primary coolant system. For the SG nozzle-to-safe end weld (which has Inconel 82/182 weld metal, known to be susceptible to PWSCC) the inside bore of the nozzle buttering and the Inconel 82/182 weld is clad using Inconel 52/152 weld metal which is resistant to PWSCC. Since Inconel 82/182 weld material is not exposed to primary coolant, PWSCC should not be a concern for this weld.

The NRC staff confirms the licensee's conclusion that the primary loop piping can be shown to exhibit LBB behavior consistent with the guidance in draft SRP 3.6.3 and NUREG-1061, Volume 3. This conclusion is based on the licensee's margins on leak rate, flaw size and combination of loads for stability of crack. The licensee's RCS leakage detection system is capable of detecting a leakage of 1 gpm in four hours. Based upon this information, the NRC staff concludes that LBB behavior has been demonstrated for the primary loop piping at PBNP, Units 1 and 2.

The NRC staff reviewed the licensee's submittal of the analyses of the LBB evaluation for the primary loop piping for PBNP, Units 1 and 2. Because acceptable margins on leakage and crack size have been demonstrated, the NRC staff concludes that the primary loop piping will exhibit LBB behavior for the power uprate and for the period of extended operation. Furthermore, the NRC staff concludes that the licensee should be permitted to credit this conclusion for eliminating the dynamic effects associated with the postulated rupture of primary loop piping from the structural design basis of PBNP, Units 1 and 2, consistent with the provisions of 10 CFR Part 50, Appendix A, GDC 4.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Wisconsin State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

These amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or change a surveillance requirement. The staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluent that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously published a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding (70 FR 6466). Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: P. Patnaik

Date: June 6, 2005

cc:

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